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FRESH FUEL VERIFICATION  
FEASIBILITY STUDY

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## FRESH FUEL VERIFICATION FEASIBILITY STUDY

### ABSTRACT

The study examined several methods of verifying crates of fresh CANDU fuel without opening the crates. Of these methods only a passive gamma technique and an active neutron technique were chosen as being appropriate. Computer simulations supported by physical measurements were performed to determine the applicability of various detectors, detector configurations, detector locations, etc. The results indicate that for the two techniques chosen a crate of fuel can be verified without its being opened provided modifications are made to the crate to permit a detector or source probes to be inserted. Estimates of the cost of modifying existing crates and procuring the measurement equipment are provided.

### RÉSUMÉ

Le rapport examine quelques méthodes pour vérifier les caisses de combustible CANDU, sans les ouvrir, et conclut que deux méthodes seulement sont appropriées : une technique passive faisant appel aux rayonnements gamma et une technique active ayant recours à une source de neutrons. Des simulations ont été effectuées sur ordinateur à partir de mesures physiques pour déterminer l'applicabilité des divers détecteurs, de leur forme, de leur emplacement, etc. Selon les résultats, tout indique que les caisses de combustible peuvent être vérifiées sans être ouvertes, en utilisant les deux techniques retenues, pourvu que les caisses soient modifiées afin qu'un détecteur ou une sonde neutronique puisse y être introduit. Le coût estimatif des modifications des caisses existantes et de l'approvisionnement en matériel de mesure est indiqué.

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Executive Summary

Fresh Fuel Verification Feasibility Study

This feasibility study examines several methods for non-intrusively verifying crates of fresh fuel for CANDU reactors. The techniques that were considered included weight measurements, use of RF probes (metal detectors), use of microwave reflectometers, gamma techniques, and neutron techniques. For various reasons, all but the passive gamma technique and active neutron technique were discarded as being inappropriate. Further study of these techniques involved use of computer simulations to predict the applicability of using the techniques under various detector configurations as well as to predict performance under differing diversion scenarios. The results of the computer simulations were verified by actual physical measurements of a crate of fresh fuel in which single bundles could be removed from different locations.

The results of the study indicate that for either technique, the use of external sources and detectors will not provide useful results in the verification of crates of fresh fuel. The study indicates, however, that modifications could be made to the design of the packaging crate that would enable a crate of fresh fuel to be verified without opening the crate. However, because of stacking arrangements at facilities, the removal of crates from storage positions will likely be required. The time required to set up and verify a full fuel crate is estimated to take from 15 to 20 minutes.

### Gamma Techniques

The study proposes two detector arrangements which could be used for the verification of crates of fresh fuel. The first alternative is to place a detector in an interstitial position so that the detector verifies four bundles simultaneously. This detector will be able to detect if one or more of the four bundles is missing, though exactly which bundle position is vacant cannot be determined.

A second approach is to use a detector that is shielded so that each individual bundle can be viewed. This approach results in an ability to verify the presence of a specific bundle. The drawbacks associated with this technique are related to the fact that each bundle must be individually counted, resulting in a long verification time if an entire crate is to be verified. To overcome this, the report proposes the use of a multidetector probe. Such a probe was not built during the feasibility study. This proposal results in a significant increase in cost to implement this technique in comparison to the other proposals.

The results of the study indicate that for the gamma technique, a single bundle deficiency in a crate of fuel will produce a deviation from a measurement on a full crate of 10% in the worst case and 25% in the best case.

### Active Neutron Technique

The active neutron technique is based on the measurement of isotope produced high-energy neutrons which have been scattered by the fuel bundles. In this technique, neutrons are released into the interior of the fuel crate, and the resultant neutron flux is measured on the crate exterior. Any deviations in this measured flux which exceed a predetermined limit can be taken as a deficiency of one or more fuel bundles in the crate.

The results of the study indicate that the source must be inserted into the crate and that by using a detector exterior to the crate, the presence of groups of six bundles can be verified. The location of the missing bundle cannot be determined. Counting times of the neutron technique are comparable to those of the gamma technique. A missing bundle results in neutron flux deviations in the order of 10% in the worst case and 50% in the best case.

One disadvantage of the active neutron technique is that a neutron source is required. This study recommends that an Am-Be source of 10 Curies would be appropriate for using this technique. Use of the neutron source would about double the potential dose that an inspector would receive from using the passive gamma technique. The health physics concerns are only marginal with use of the neutron source. However, it is noted that there may be some difficulty in shipping sources from country to country.

Comparison of the Gamma and Neutron Techniques

A short comparison of the gamma and neutron techniques are listed in Table 1. A more detailed comparison is given on page 104 of the report.

Table 1: Various Aspects of the Verification Techniques Studied

|                                                             | Neutron               | 4 Bundle/Detector<br>(Gamma Technique I) | 1 Bundle/Detector<br>(Gamma Technique II) |
|-------------------------------------------------------------|-----------------------|------------------------------------------|-------------------------------------------|
| Crate Disassembly<br>Required                               | No                    | No                                       | No                                        |
| Substitute<br>Material Detection<br>Ability                 | Yes                   | Yes                                      | Yes                                       |
| Total Number of<br>Measurements/Crate<br>Using One Detector | 12                    | 12                                       | 36                                        |
| Counting Time Per<br>Measurement                            | 120 SEC<br>@ 2% Error | 20 SEC<br>@ 2% Error                     | 40 SEC<br>@ 2% Error                      |
| Non-Standard<br>Equipment                                   | None                  | Detector Probe                           | Detector Probe                            |
| Setup Time                                                  | 300 SEC               | 300 SEC                                  | 300 SEC                                   |

As noted previously, both techniques require that the fresh fuel crate being verified must be penetrated by some type of probe. This requires that the several modifications be made to the existing bundle packing crate design and that some type of probe be fabricated. It is estimated that changes in the fuel crate design and tooling will cost an estimated \$8,000.00 for the majority of manufacturers and up to \$30,000.00 for one manufacturer who currently uses a slightly different fuel crate design not suitable for the proposed verification techniques.

Besides the probes which must be supplied, several pieces of equipment are required to implement each technique. These pieces of equipment are already contained in the IAEA's inventory of NDA equipment. The proposed configuration for the equipment for the verification of spent fuel are shown in Figure 1.

The estimated cost to implement the two techniques varies upon how fast the system must characterize the full case. In a case where a full crate has to be verified in a short time, the basic technique is modified to incorporate the parallel measurements, thereby reducing the overall measurement time. The anticipated cost of implementing the various techniques are shown below. In the case of Gamma Technique II, the price is based on a one detector assembly which contains 4 detectors, (1 for each bundle). In the other two techniques, the prices are based on the use of detectors and support equipment in a quantity sufficient to allow two parallel measurements to be performed at one time. In

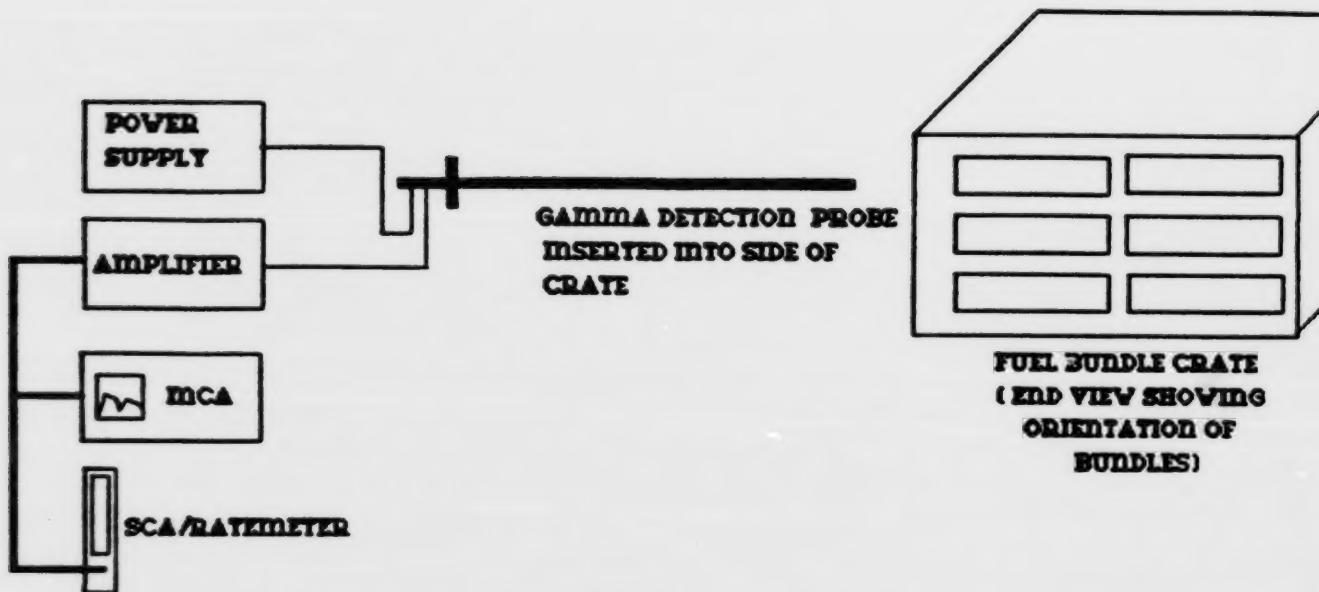
| <u>Technique</u>   | <u>Cost</u> | <u>IAEA Incremental Cost</u> |
|--------------------|-------------|------------------------------|
| Gamma Technique I  | \$21,000.00 | \$ 7,000.00                  |
| Gamma Technique II | \$27,000.00 | \$15,000.00                  |
| Neutron Technique  | \$18,000.00 | \$ 9,000.00                  |

the case of the gamma I technique, this configuration translates into a single probe with two detectors, each of which are capable of monitoring 4 bundles. An example of the neutron technique is shown on page x.

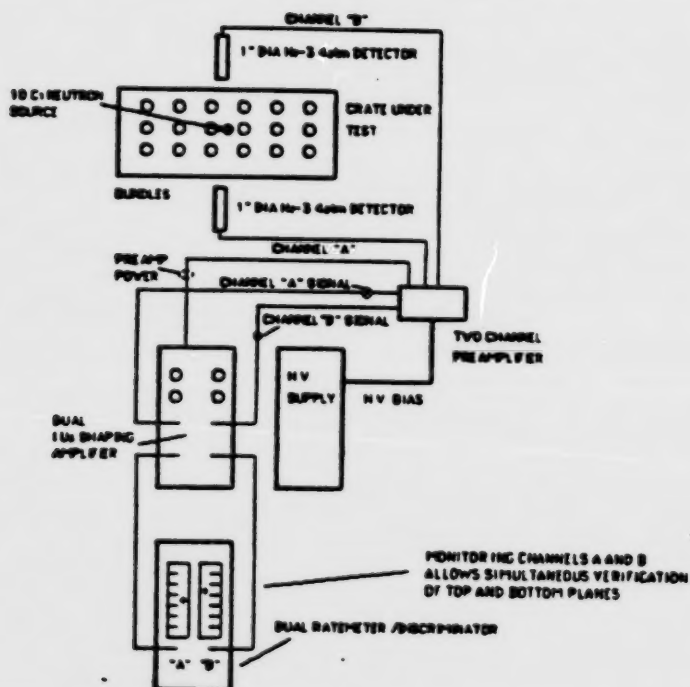
The incremental cost is that cost which would be incurred excluding equipment from IAEA inventory. The feasibility study indicates that it should be possible to verify a crate of fresh fuel with any of the techniques recommended. In order to implement a technique, the Agency must assemble the appropriate equipment and manufacture the appropriate probes. In addition, the design of the fresh fuel packing crate must be modified.

In order to establish procedures that would be used by inspectors, the IAEA must determine the go/no go conditions by which an inspector would determine the presence or absence of fuel in the crate. This would involve taking a number of measurements on full crate of fuel to obtain baseline information and then obtain similar data for fuel crates having one or more bundles missing.

**FIGURE 1. PROPOSED VERIFICATION TECHNIQUES**



**GAMMA BASED TECHNIQUE - EQUIPMENT SETUP**



**NEUTRON BASED TECHNIQUE - EQUIPMENT SETUP**



## 2.0 IDENTIFICATION OF FRESH FUEL VERIFICATION PROBLEM

There is a need to verify fresh fuel stored at nuclear reactors and fuel fabrication plants. In the Candu fuel cycle the majority of fresh fuel is stored in packing crates located in racks or stacked in rows several crates high. Moving crates of fuel and opening them to verify their contents is time consuming and increases the probability of personal injury to the inspectors and damaging to the fuel, both of which can result in considerable expense to the fuel owner. A more desirable means of verification would be the non intrusive assay of the fuel crate contents. Since no technique has yet been developed to perform verification via these means this feasibility study has been carried out.

### 2.1 OVERVIEW OF VERIFICATION PARAMETERS

To gain some insight into how Candu fuel is fabricated and shipped, consider the basic fuel element used in the Candu system. In the most simple terms, Candu fuel consists of a number of small diameter natural uranium dioxide fuel pellets configured as zirconium clad pencils. Up to 38 pencils are then radially arranged to form a four inch diameter "bundle". These bundles are then used as the reactor fuel. Once fabricated, the bundles are shipped from the fabrication site to the nuclear plant in wood, plastic and paper crates, each of which contains 36 bundles. These crates then remain in storage until required for reactor refueling, at which time the fuel is removed and the crate discarded.

Since fresh fuel is stored in crates during most of its preirradiated

life the inspection of those crates becomes a key element of any fuel cycle verification process. As it is well known that fresh CANDU fuel is relatively safe in terms of toxilogical chemical and radiation effects it would be expected that the task of packaged fresh fuel verification (FFV) would be quite simple. One would reasonably expect that the task of verification could consist of unpacking each fuel crate, visually inspecting each fuel bundle to verify mechanical integrity, verifying the bundle contents through various means and then repacking the fuel bundles in the shipping crate. This could be repeated for each fuel crate in the nuclear facility. Unfortunately this process would be very time consuming as each facility can store up to a few hundred crates of fuel. To reduce the time required to carry out this procedure the authorities responsible for FFV typically adopt a statistical sampling approach to the crate examination. This technique results in a smaller number of samples being taken.

While the statistical sampling approach has reduced the time required to complete the FFV procedure the technique used for actual verification is still based on mechanical manipulation of the fuel. Even though manual inspection is quite reliable in terms of fuel verification accuracy, it must be remembered that each crate consists of 36 heavy (50 pounds) and difficult to handle fuel bundles and that any mechanical shock to the fuel (ie. dropping on floor) could result in failure of the fuel while the fuel is in the reactor. As a result, any signs of mechanical fuel damage will cause the fuel to be rejected by the reactor fueling staff. In recognition of these inherent problems it would be highly desirable to develop an FFV technique that does not require the fuel to be mechanically handled.

The remainder of this report addresses the feasibility of using non mechanically based FFV techniques. The key requirement that was stressed during

the investigation of these techniques is that the FFV procedure be executable without disturbing the internal integrity of the fuel crate. Furthermore it was considered a requirement that only minimum changes in the existing fresh fuel storage, shipping and packaging be needed for implementation of the new technique.

This report is separated into several parts, each part deals with a particular aspect of the proposed FFV techniques. The final section of this report presents a relative comparison between the techniques judged to be best suited for the job of on site FFV.

### 3.0 CHARACTERIZATION OF FUEL STORAGE FACILITIES

#### 3.1 FUEL STORAGE FACILITIES

In order to develop a better FFV technique it was required that a comprehensive study of the existing fresh fuel storage environment be carried out. To these ends an investigation into typical fuel storage arrangements was undertaken via on site measurements at both Ontario Hydro's Pickering Nuclear Generating Station (PNGS) and Pointe Lepreau NGS (PLNGS). On site data was collected regarding mechanical layout (ie. crate to crate spacing) and radiological background conditions.

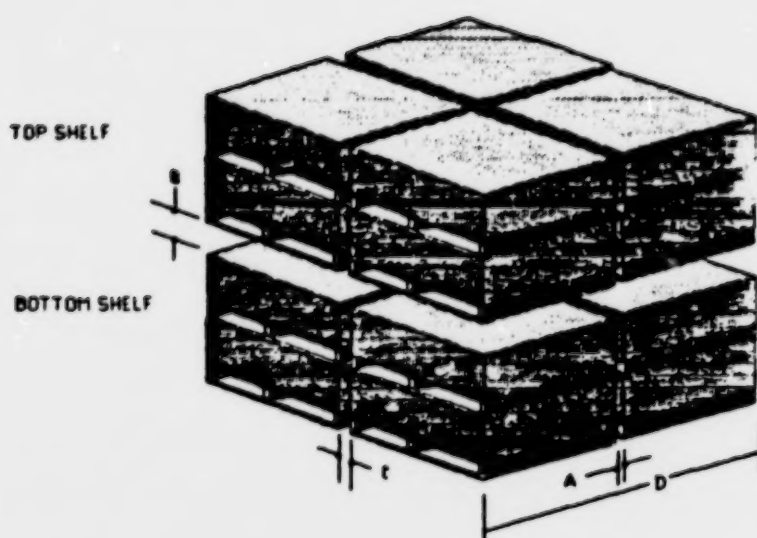
##### 3.1.1 MECHANICAL LAYOUT

On site inspection revealed that the mechanical layout employed by the various facilities were essentially the same. The only differences between the storage sites were minor variations in crate to crate spacing and overall height of the storage racks. In all cases, the fuel storage room consisted of an access corridor (which can be accessed by a fork lift truck) lined with adjustable racks. The fuel is stored end to end along these racks with a maximum stacking of two crates. This arrangement is shown in Figure 1. From this figure it will be noted that the fuel crates never rest on the floor. Also it will be noted that each shelf contains crates stacked 2 high.

Since it was decided at an early stage that it would be desirable to have the option of executing the FFV procedure while fuel is still on the storage rack, measurements of minimum clearances around each fuel crate were

made. During the course of these measurements it was found that crate to crate spacing is not currently considered as a critical factor. As a consequence, lateral (side to side) intercrate spacing was found to vary from 0 to 1 ft. It should be noted that although the intercrate spacing in the existing storage arrangement is uncontrolled, the current crate and storage shelf dimensions would allow up to 4" between each crate. This value was found to be invariant between each site. It was also determined the vertical crate spacing was in the worst situation only 1 inch. Through examination of on site equipment it was discovered that spacing could be easily varied in 1 inch increments.

To summarize the on site mechanical parameters in a concise manner Table 3.1.1 has been prepared. Note that this table presents data from both facilities. Examination of the orientation and packing density of the fuel bundles has shown that the fuel bundles are stored parallel to the access corridor. Because of this packing configuration, it is only possible to obtain an unobstructed view of the sides of 6 of the 36 fuel bundles. All other bundles are obscured by these first 6 bundles. The impact that this storage configuration has on the implementation of the proposed nonintrusive FFV technique is covered in the section detailing gamma technique implementation.



A: CRATE TO CRATE LATERAL SPACING  
B: SHELF TO TOP OF CRATE CLEARANCE  
C: CRATE TO CRATE HORIZONTAL CLEARANCE  
D: DEPTH OF STACKING

FIGURE 1 - FUEL CRATE SPACING

Table 3.1.1 - Fuel Crate Clearances  
All dimensions refer to Figure 1

| Measurement                                          | Maximum          | Minimum        |
|------------------------------------------------------|------------------|----------------|
| Crate to crate lateral clearance<br>(Dimension A)    | 6 in. (PLNGS)    | 0 in. (PNGS)   |
| Shelf to top of crate clearance<br>(Dimension B)     | 5 ft. (PLNGS)    | 1 in.(PNGS)    |
| Crate to crate horizontal clearance<br>(Dimension C) | 1 ft.            | 0 ft.          |
| Depth of stacking<br>(Dimension D)                   | 2 crates (PLNGS) | 1 crate (PNGS) |
| Height of stacking                                   | 2                | --             |
| Number of shelves                                    | 4 (PNGS)         | 2 (PLNGS)      |

### 3.1.2 ON SITE RADIATION MEASUREMENTS

As noted previously, a survey was completed at both NGS sites to determine the magnitude and spatial distributions of radiation backgrounds in the fresh fuel storage areas. This survey was undertaken in order to determine what, if any radiation fields existed. Special emphasis was placed on determining whether or not the magnitude of the observed fields were sufficient to corrupt the readings obtained from proposed radiation based FFV techniques. Since the techniques investigated for FFV include both neutron and gamma radiation, a survey was undertaken for both these radiation types.

The format used in the radiation survey was the point wise measurement of flux at a variety of positions in the fuel storage room. Measurements were taken at a minimum of three different elevations for each position. This procedure was carried out with both gamma and neutron detecting equipment. The detailed gamma flux mapping of the storage room was carried out with an energy compensated Gieger tube based counter. Once the energy insensitive survey was completed, several randomly positioned samples were taken with a 2x2 inch NaI(Tl) detector connected to a Multi channel analyser (MCA).

Detailed mapping of the neutron flux in the fuel storage area were taken with both a He3 neutron detector moderated with 6 inches of high density polyethylene and a Long type dosimeter provided by the station. Both of these counter configurations are known to provide high counting efficiency for the range of neutron energies of interest for the proposed technique (0 to 5 MeV).

The results for both the gamma and neutron surveys are shown in Table 3.1.2. From these figures it becomes immediately obvious that there is no measurable neutron flux in the confines of the fresh fuel storage rooms at

either site. It will also be noted that gamma fields in the fresh fuel storage rooms are dependent on both the vertical and horizontal position. These observations are quite important as it indicates that a gamma based FFV technique must contend with variable background gamma fields whereas a neutron based technique will be free from background interference.

Table 3.12- Background Radiation Levels in Various Fresh Fuel Storage Facilities

| Location                                | Max.Gamma | Min.Gamma | Max.Neutron | Min.Neutron |
|-----------------------------------------|-----------|-----------|-------------|-------------|
| 1. Auxiliary fresh fuel room            | 7 mR/hr   | 1 mR/hr   | 0           | 0           |
| 2. Rejected fresh fuel room             | 7 mR/hr   | .2 mR/hr  | 0           | 0           |
| 3. Primary fresh fuel storage room East | 10 mR/hr  | 2 mR/hr   | 0           | 0           |
| 4. Primary fresh fuel storage room West | 10 mR/hr  | 3 mR/hr   | 0           | 0           |
| 5. PLNGS fresh fuel storage room East   | 7 mR/hr   | 2 mR/hr   | 0           | 0           |
| 6. PLNGS fresh fuel storage room West   | 6 mR/hr   | 3 mR/hr   | 0           | 0           |
| 7. Fuel storage shelf                   | 10 mR/hr  | 7 mR/hr   | 0           | 0           |

Unless otherwise noted, all measurements were taken at PNGS.

### 3.1.3 DISCUSSIONS WITH ON SITE PERSONNEL

In addition to the surveys that were carried out on site, discussions were held with the the relevant fuel handling personnel. These discussions dealt with the types of mechanical manipulation that would be acceptable for periodic FFV. Topics discussed included what would constitute an acceptable fuel handling procedure as well as what time and manpower commitments could be made during a fresh fuel inspection. The conclusions reached from these discussions were then used to formulate guidelines for the development of the operating procedures for a new FFV technique.

The responsibility of fuel management and control rests with the fuel handling group. The primary concern of this group is, in order of precedence, worker safety, fuel quality control and maintainence of a proper core fueling schedule. In regards to fuel quality control it is the opinion of station staff at both NGS sites that the unnecessary handling of fresh fuel is to be discouraged.

If proper handling of the fuel does take place, there is the possibility of damaging the fuel. In a worst case, a bundle would be damaged even though the damage would not be readily observable. In this case, the damage could potentially cause fuel failure while the fuel is in use. This would be undesirable since costs will be incurred by the utility through higher dose to maintainance personal and through added clean up costs related to heat transport system decontamination.

With regards to worker safety, concerns were voiced regarding the safe handling of fresh fuel. The major concern was that the handling of fresh fuel bundles can pose a back injury threat to both the IAEA staff as well as the

fueling operators assisting in the FFV task. The primary threat in this case is the one associated with handling of a large quantity of fifty pound bundles without the assistance of proper equipment. Furthermore, at one NGS site, limitations imposed by insurance coverage exempts any injury compensation for back injuries that result from lifting more than 40 lbs. As a result of these concerns, the manual packing and unpacking of fresh fuel is considered to be an undesirable means of accomplishing FFV.

The above mentioned discussions clearly indicate that the manual disassembly and unpacking of fresh fuel crates is not considered as being an ideal method of accomplishing FFV. As a result, the only apparent solution is to develop an FFV method that can be applied from the outside of a fresh fuel crate.

Discussions were also held in regards to the development of a technique suitable for use with fuel stored on the storage rack. The PLNGS staff considered this type of technique to be desirable. In order to allow this type of testing to take place, it was also indicated that it may be feasible to provide a clearance of up to four inches between crates. It was noted that this action would only require that the support arms be placed in the appropriate slot (which are 1" apart).

In the case of PNGS staff, concerns were raised with regards to the safety of implementing an on the shelf testing technique. The primary concern was related to the fact that fuel stored on the top shelf of the PNGS fresh fuel room is well out of reach of a person standing on the floor. As a result, proper scaffolding and safety harnesses would have to worn by the persons executing the FFV procedure. In recognition of this potential safety hazard it was indicated that it would be more desirable to move the fuel from the storage

rack to the floor with a fork lift prior to execution of the FFV procedure. It was felt that this maneuver would not cause disruption of the stations daily work schedule since a forklift is already available for the purpose moving the fuel storage crates.

Finally, discussions were held in regards to the availability of a 117V AC power source that could be used to power the equipment needed for FFV. The outcome of this discussion was to find that while 117V is available in the form of twist-lock plugs, however a preferable approach would be to use battery operated equipment during FFV.

To summarize the findings of the discussion with on site personal we have the following:

1. Unpacking of the fuel is considered undesirable because of operations and safety concerns.
2. Movement of the fuel crates with a fork lift is acceptable during FFV.
3. On shelf testing of the fuel would be acceptable for fuel stored on the lower shelves whereas fuel on the higher shelves should be moved to ground level with a fork lift for the duration of the FFV procedure.
4. 117V AC power is available to power any measurement equipment required for FFV but the use of Battery powered equipment is preferable.

### 3.2 CHARACTERIZATION OF FRESH FUEL PARAMETERS

In order to develop an accurate method for FFV, a set of data related to the physical properties of fresh fuel was compiled. This data is presented in tabular form in table 3.2.

Table 3.2 - Fuel Bundle Data

|                                         |   |                                                   |
|-----------------------------------------|---|---------------------------------------------------|
| Length of bundle                        | - | 49.5 cm                                           |
| O.D. of bundle                          | - | 10.2 cm                                           |
| Weight of bundle                        | - | 23.7 kg                                           |
| Sheath thickness                        | - | .041 cm                                           |
| Minimum inter-element spacing           | - | .127 cm                                           |
| UO <sub>2</sub> diameter                | - | 1.433 cm                                          |
| Maximum UO <sub>2</sub> orthogonal area | - | 45.29 cm sq.                                      |
| U-235 content                           | - | .14 kg                                            |
| U-238 content                           | - | 21 kg                                             |
| Photon Equivalent Specific activity     | - | 10,000 Photons/sec/cm sq                          |
| Characteristic gamma emission           | - | U-235- many but primary peak at 185.7 KeV         |
|                                         | - | U-238- broad spectrum Xray no characteristic peak |

#### 4.0 FRESH FUEL VERIFICATION TECHNIQUE DEVELOPMENT PHILOSOPHY

#### 4.1 OVERVIEW OF POSSIBLE FRESH FUEL VERIFICATION TECHNIQUES

Several techniques for the non intrusive inspection of the fresh fuel crates were studied. Each of these techniques were based on a measureable physical property of the fuel contained inside of the shipping crate. Table 4.1 lists the various techniques that were studied along with the corresponding physical property being measured.

TABLE 4.1 - Possible Verification Techniques

| NO. | TECHNIQUE                 | MEASUREMENT                       | COMMENTS                                        |
|-----|---------------------------|-----------------------------------|-------------------------------------------------|
| 1.  | Weight                    | mass of fuel                      | easily masked                                   |
| 2.  | RF Probe (metal Detector) | Magnetic Property if fuel bundles | not specific to UO <sub>2</sub> , can be masked |
| 3.  | Microwave Reflectometer   | position of bundles               | no measure of bundle contents                   |
| 3.  | Gamma monitoring          | natural activity of uranium       | isotope specific measurement possible           |
| 4.  | Neutron monitoring        | neutron cross-section of fuel     | Requires radioactive source for test            |

As a result of this examination several techniques were ruled out due to major shortcomings. Of particular interest was the fact that the more simple techniques were easily masked, with the result that they were ruled out as candidates. Techniques falling under this category were: weight, RF probe and reflectometer. From the table one will find that each of these techniques, while capable of providing limited data about a sealed fuel crate, has a major flaw that adversely affects its credibility or confidence level when used in a

verification scheme.

The remaining techniques, (gamma monitoring and neutron monitoring) while being slightly more difficult to implement have been found to be capable of providing measurements that can be used to derive a reasonably accurate measure of the contents of a sealed fuel crate. The remainder of this report details the physics and practical aspects of implementing each of these techniques.

#### 4.2 GAMMA BASED FFV DEVELOPMENT PHILOSOPHY

After preliminary investigations it was determined that fresh fuel verification could most easily be accomplished through nuclear means. By using the past experience of the research team it was determined that passive gamma and active neutron assay would be the most likely candidates for successful development into a new FFV method for the Candu system.

Since passive gamma based assay is widely used in verification of enriched fresh fuel, its application to the problem of natural uranium FFV will be investigated first. As natural uranium undergoes radioactive decay, photons and particles of specific energies are emitted. The photon portion of this decay behaves much like light photons emitted from a light source. It is in theory possible to locate the position of the radioactive emission by observing and mapping this photon flux in the vicinity of the emitter, that is provided that the view of the photon emitter is unobstructed.

Application of the mapping technique as a means of locating fuel bundle position (and therefore total number of bundles) in a sealed fuel crate would

require that the uranium emit a sufficient number of photons to permit easy observation and that the crate construction permits a fairly unobstructed "view" of the radiation emitted from the bundles. Investigation reveals that in the case of refined natural uranium the main radioactive components are U-235 and U-238. These isotopes emit a range of energies with U-235 producing roughly one characteristic 185.7 Kev for every two decays. U-238 does produce a weak characteristic gamma (50 KeV), but this is not easily observable. Fortunately, U-238 does emit a fairly large energy flux that can be used for the detection of the bundle. If we examine the fuel crate construction materials we find that the crate is constructed primarily of low density hydrogenous material (polystyrene) and paper. Both these materials exhibit very low macroscopic gamma cross-sections at the energies in question and may therefore be considered as transparent. Finally, if we examine the crate packing configuration we find that several of the fuel bundles in the center of the crate are effectively shielded from view by surrounding bundles. As will be shown latter, this creates problems when the mapping technique is to be implemented.

The next factor to be considered in the application of gamma imaging to FFV is the bundle to bundle interference. In attempting to apply a gamma technique to view the interior of a fuel crate, we find that even though fuel bundles are detectable via their own emitted gamma photons, they also have a very high gamma absorption cross section. This high cross section makes the bundles opaque as far as uranium produced photons are concerned. As a consequence, we find that fuel bundles will tend to shield themselves as well as each other in a way that is analogous to the shielding of light by an opaque object. Due to the high density of bundles and the packing arrangement used in the fresh fuel crate it is very difficult to obtain a clear view of each bundle,

particularly those in the center plane of the crate. In recognition of this problem, a mandatory requirement of the gamma imaging technique development program is the development of a method to see as many and as much of the fuel bundles as possible without disturbing the integrity or design of the shipping crate.

If we now expand our investigation to the exterior of the fresh fuel crate, we find that the crate undergoing verification is surrounded by a large number of fresh fuel crates. This becomes a problem since the radiation being emitted from these crates is identical to that being measured. It is therefore highly probable that this exterior radiation can contaminate the measurement being made on the crate in question. In fact, the on site data collected during the radiation survey indicated that the background flux levels produced by these crates approach the flux levels measured on contact with the fuel crate.

Finally, we turn our attention to the question of enrichment verification. The ability to obtain an enrichment figure for the fresh fuel is dependent on the fact that natural uranium contains two main radioactive components (U-235, U-238). Fortunately the ratio of the concentration of these components provides a direct measure of enrichment. Since these two isotopes emit gamma photons of different energies, measurement of the emission rates at the two energies can provide a direct measure of enrichment status. It should be noted that this technique is a simple extension of the FFV procedures noted previously. Also, it should be noted that the technique of energy selective counting for enrichment determination is widely accepted. Although widely used in enriched fuel situations, its application to FFV of crated natural uranium fuel must be further studied allow the development of an optimal measurement procedure.

To better understand the principles of gamma based FFV, a tutorial on the subject has been provided in Appendix 1.

#### 4.3 NEUTRON BASED FFV DEVELOPMENT PHILOSOPHY

Since it was not known whether the problems inherent in the gamma based FFV approach were resolvable through a short term development program, a second more advanced technique was also investigated. In this technique, neutrons of various energies are used to probe the nuclear characteristics of the fuel components. In order to understand how this technique works and can best be applied to FFV, one must review the basic mechanics of neutrons and their interactions with various materials.

As a first step, one will recall that neutrons will, to first approximation, undergo two types of interaction with materials. The first of these interactions is scattering. In this case the neutron bounces off of the nucleus contained in a target material. This process can be energy conserving (elastic) or energy consuming (inelastic) dependent on the type of nucleus in the target. The second type of interaction is absorption. In this interaction the target nucleus absorbs the incident neutron. After absorbing the neutron the target can either be permanently changed to a new isotope or it can become unstable and decay to a different isotope or element.

In order to apply the physical properties of neutron/nucleus interaction to FFV we must first set up a geometry in which the fuel under test is irradiated with neutrons from a source. Currently there are a large variety of sources available, each with a specific emission energy spectrum. The details on how an optimum source is to be selected therefore becomes one of the goals of

a development program.

Once a source is selected it would be set in a geometry so as to irradiate the fresh fuel in a desired fashion. This is readily accomplished by placing the source adjacent to a single or group of fuel bundles via manual means. Once the irradiation of the fuel has started the previously mentioned interactions will occur at varying rates. In particular we find that the uranium dioxide will both scatter and absorb the incident neutrons. In the case of absorption we find that two interactions occur. The first case is simple absorption where the neutron and its kinetic energy is totally absorbed. In the second case we have the phenomenon of fission where the uranium nucleus is caused to fission. This latter effect can then be subdivided into two categories. The first category being thermal fission, an effect where the U235 fissions as a result of the absorption of a low energy neutron. The second category corresponds to fast fission where U238 is caused to fission as a result of absorption of high energy neutrons. It should be noted that the magnitude of both the absorption and scattering effects described above are particular to each isotope, thus few if any elements possess the same absorption to scattering relationship. This nucleus specific phenomenon can therefore be used as a signature from which to determine if the material being irradiated is actually the material expected.

Once it is acknowledged that each element will display different nuclear characteristics, the problem digresses to one of how neutron detection and characterization will take place. In an ideal situation one would like to record all parameters that relate the movement of an emerging neutron. To do this requires that we obtain vectored velocity data. Unfortunately, collection of this type of data is complex and very inefficient. This high cost and lack

of efficiency rules out the use of such a scheme. If we then examine the next preferable method of data collection we enter the realm of energy spectroscopy. Investigation into the equipment needed to collect a neutron energy spectrum will reveal that the collection process is both more efficient and less complicated than that required for neutron velocity acquisition. Despite this improvement there still exist definite financial and technical costs associated with collection of this type of data. If, in light of these known factors we digress to the next lower level of sophistication we reach the most simple of neutron devices, the neutron counter. In this simple collection scheme, only the presence of the neutron is detected, with no regards to neutron energy and direction. The advantages gained through the use of such a scheme are significant in terms of over all neutron economy and cost savings.

To summarize, we find that we must optimally determine source position, source energy spectrum and detector characteristics if we are to successfully exploit a neutron based FFV technique. Since the variation of a single parameter will interact with the two remaining parameters, the development program must be able to contend with simultaneous optimization of three terms. The methodology as well as exact principles used to select these terms are given in sections that follow.

## 5.0 PASSIVE GAMMA BASED ASSAY TECHNIQUE

This section of the reports deals with the technical and logistical considerations of implementing a passive gamma based FFV technique. As noted previously, a tutorial on the principles of gamma imaging is given in appendix 1.

### 5.1 SIMULATION

Since the discussions relating to gamma based fresh fuel verification have up to this point been based on qualitative properties, a computer based parametric model of the gamma based verification schemes was developed. By adopting a computer based development method, the time consuming and error prone task of manual measurement has been simplified, thereby enabling much faster verification and optimization of the technique under study.

In order to undertake the computer study, two general purpose gamma flux simulation programs were developed, tested, and run with varying geometric conditions.

The first of the programs to be developed was a spatial gamma flux prediction program for prediction of the gamma flux produced by fresh Candu fuel bundles. The algorithm for this program (called BPGF for Bundle Produced Gamma Flux predictor) is based on a simple ray tracing and minimum angle scattering model. With this algorithm, numerical values are derived for flux profiles of the unscattered UO<sub>2</sub> emissions.

The assumptions used in the simulation model are as follows:

1. Classical exponential attenuation is assumed.
2. Scattering considered possible for angles less than  $10^\circ$  degrees.  
Scatter probability defined by Klien-Nishina formula.
3. Gamma emitters simulated by discrete ordinates.
4. Collimator walls (if any) are assumed to be gamma opaque and non scattering.

By using these assumptions we find gamma photons which possess energies that are less than 90% of the initial emission peak energy are ignored. Figure 2 shows the flow chart of BPGF. As you will note, the program can be run with a variety of geometric configurations. In particular, it should be noted that fuel pencils can be removed and replaced with pencils constructed of other materials. Also, it should be noted that the program calculates gamma flux at a point which is incremented at .1 degree intervals along the radius and 1 cm along the length of the bundle. Thus, in order to predict the response of a particular detector, the spatial flux profile must be integrated over the area of the detector.

In running BPGF, a variety of bundle pencil geometries were investigated. The predicted flux profiles along with the generating pencil geometries are indicated in figure 3.

Once BPGF was completed, the results were introduced to another program designed to predict gamma flux profiles produced by the fuel crate under a variety of missing bundle configurations. This program called CPGF (Crate Produced Gamma Flux predictor) is based on an algorithm similar to that used in BPGF. In CPGF, the detector response was modified as to simulate a collimated detector. Several types of collimators were examined. A more complete treatment of the collimator design is presented in section 5.1.1. As in the case

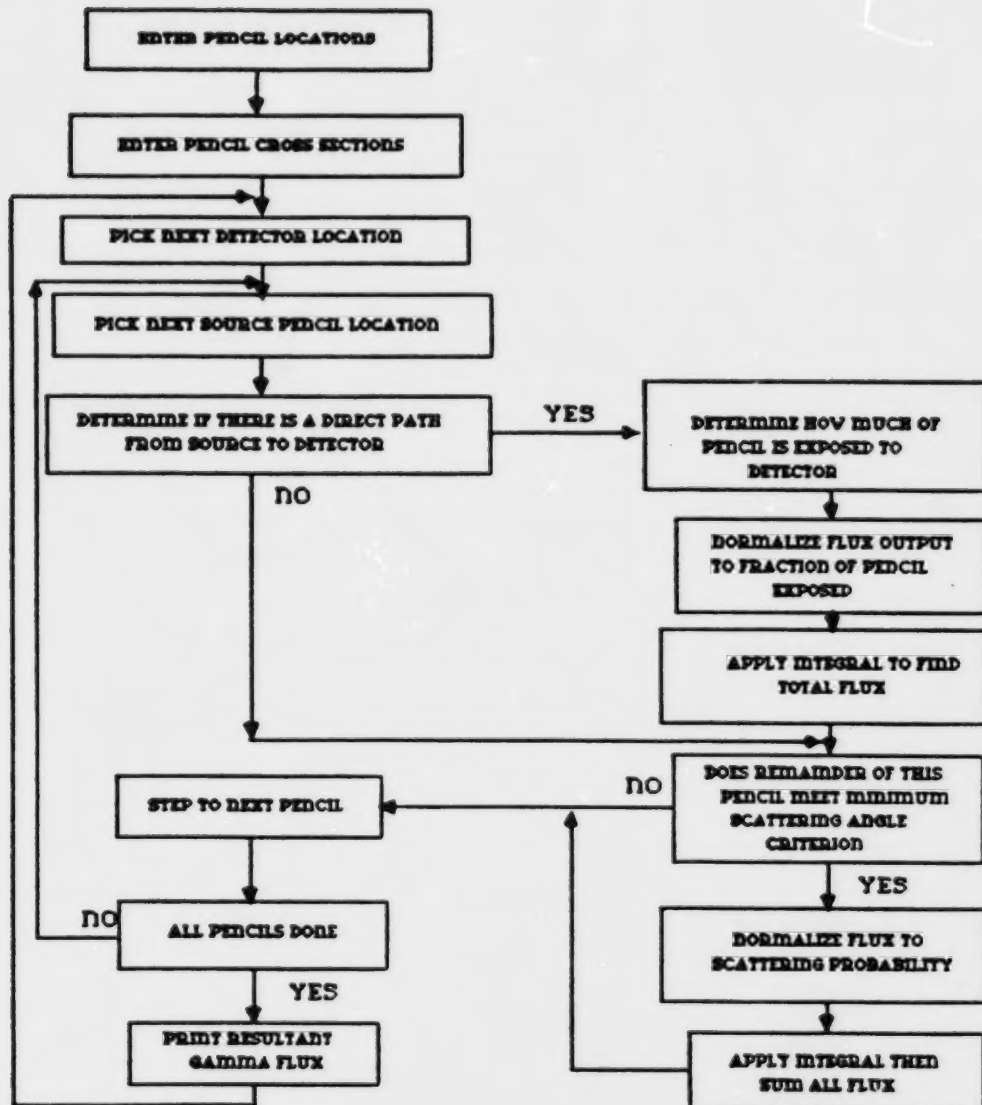


FIGURE 2 - BPGF FLOW CHART

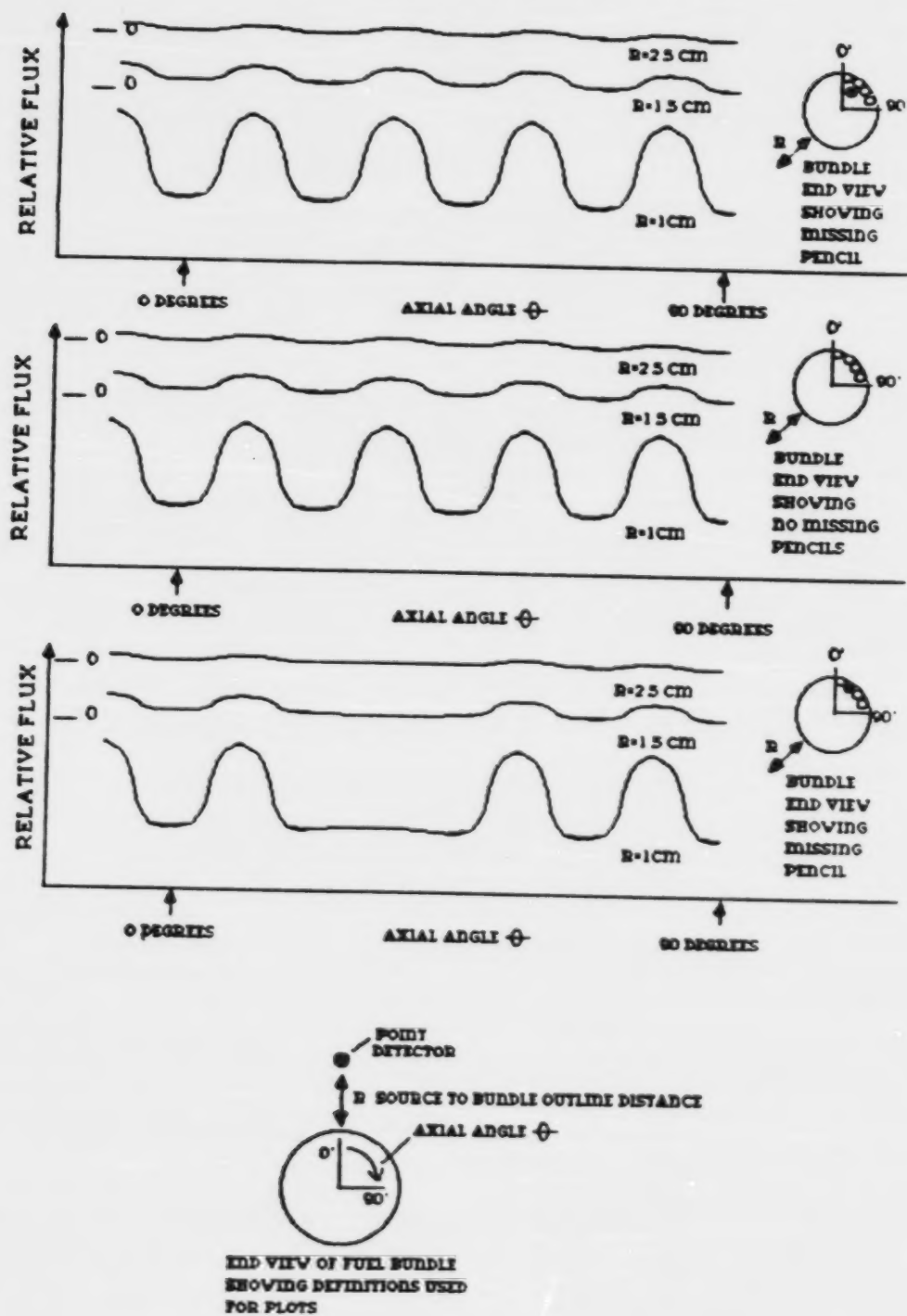


FIGURE 3-FLUX PROFILES OF ALTERED FUEL BUNDLES  
FOR SEVERAL RADII AND BUNDLE GEOMETRIES

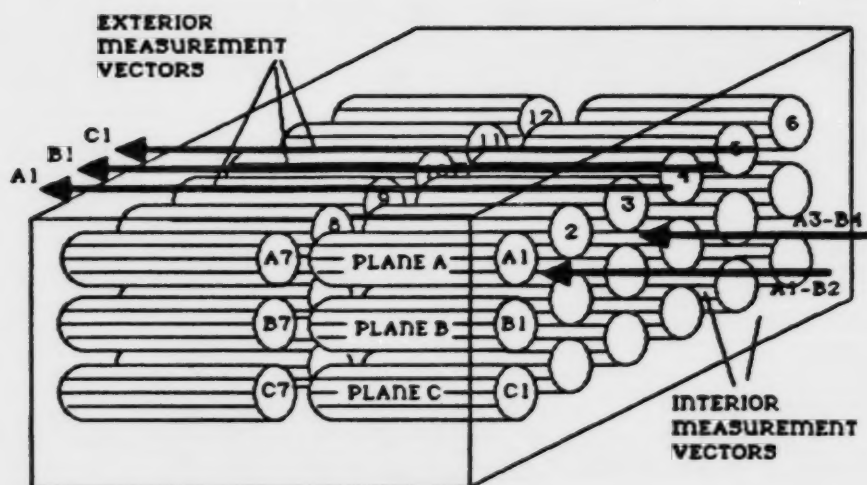
of BPGF, several assumptions were made in the design of CPGF. These assumptions consisted of the following:

1. For attenuation calculation purposes bundles were assumed to consist of a cylinder of solid UO<sub>2</sub>.
2. Radiation patterns from groups of bundles were assumed to be the summation of variously displaced gamma profiles as predicted by BPGF.
3. Symmetry was used where possible in order to minimize computation time.

In reviewing the algorithm used in CPGF it should be noted that no facility has been provided for the addition of background gamma flux. Because of this shortcoming, the results provided by BPGF must be considered as being realizable only under ideal conditions. None the less, the results from CPGF are a valid guide to the performance that can be expected from the gamma technique under ideal conditions. By using this premise, a philosophy has been adopted whereby if CPGF shows that a particular test will not work, that test will definitely not work in an in field environment. Similarly, failure of CPGF to provide adequate results will automatically nullify the need to undertake the test in the field.

Several conventions were adopted in the CPGF program. In order to visualize the process used to arrive at the various conclusions discussed in the subsequent sections, familiarization with these conventions is mandatory.

The first convention used by CPGF is the measurement direction vectors. These vectors describe the line that is traversed by the detector during a particular measurement simulation. A map of these vectors is shown in figure 4.



IDENTIFICATION OF POSITION IN FUEL CRATE IS THROUGH PLANE AND BUNDLE NUMBER.  
 -FOR MID-PLANE OR MID-BUNDLE LOCATIONS THE LOCATION IS IDENTIFIED BY GIVING THE IDENTIFICATION OF THE BUNDLE THAT SURROUNDS THE POINT IN QUESTION  
 -MEASUREMENTS TAKE PLACE ALONG THE LENGTH OF THE BUNDLES AND ARE IDENTIFIED BY THE COORDINATES OF THE CLOSEST PARALLEL BUNDLE.

#### EXAMPLES

A1 - TOP PLANE BUNDLE 1  
 B6 - MID PLANE BUNDLE 6  
 A1-B1 - BETWEEN BUNDLE A1 AND B1  
 C2-C3 - BETWEEN BUNDLE C2 AND C3  
 A1-B2 - MID PLANE IN THE CENTER OF BUNDLES A1, A2, B1, B2

FIGURE 4 - BUNDLE AND MEASUREMENT VECTOR IDENTIFICATION CHART

Referring to these vectors one will note that spatial flux profiles have been taken at positions both internal and external to the crate perimeter. The details of how these internal flux measurements were taken are given in the section dealing with on site measurements.

The second convention that is used by CPGF relates to computed flux. The flux values derived by this program are normalized to unity emission. That is to say, the total surface activity of the bundle is normalized to 1.00.

The third and last convention used relates to the flux energy. As explained in the section detailing the BPGF program the program is based on simple single energy group flux loss. As a result, only gammas whose energy falls between the full 185.7 KeV emission energy and 90% of this value are counted. This method was used both to simplify the program as well as to validate the simulation results for U-235 assay (which requires analysis of a single energy peak).

Several versions of the program were run, each with a different measurement vector. The findings that resulted from these runs are discussed in the sections that follow.

#### 5.1.1 COLLIMATOR DESIGN

As indicated in the sections dealing with the basic principles of gamma detection, we find that point gamma emitters possess a  $1/(r \times r)$  flux dependence as a function of distance from the source. Unfortunately, this equation is only valid for environments involving point gamma emitters. In the present case where we have several large fuel elements surrounding a small sized detector, the conditions required for application of the point source equation are not

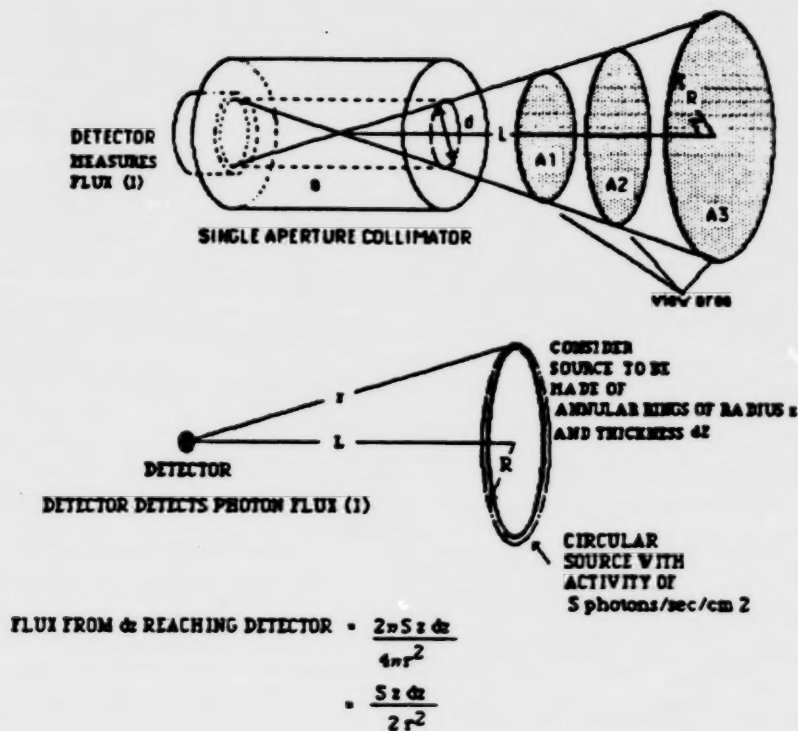
met. As a result, the computation of gamma flux in the vicinity of the detector becomes more complex. In particular, the calculation of gamma flux in this case becomes a matter of continuous integration of flux resulting from a continuum of point sources contained in an emitting body. While it is fairly simple to calculate the flux resulting from a given source geometry and activity, it is virtually impossible to derive the source geometry and activities from a single measured flux value. This one way nature of the problem necessitates measurement of a series of point fluxes in order to obtain spatial data regarding the fuel bundles.

A refinement to the single point measurement technique can drastically reduce the number of measurements that need to be taken in order to determine bundle position. Recalling the initial discussion of the topic, we find that the reliable operation of a gamma imaging technique is dependent on the measurement of the flux emerging from a small and well defined area of gamma emitting substance. To achieve this narrow view, some form of collimation must be introduced into the detector design.

The concept of a collimator is fairly simple to understand. Basically, a collimator is a gamma opaque material that restricts the view of the detector to a specific direction. The shape, density, and thickness of the collimator is chosen so as to minimize the number of internally scattered gamma photons reaching the detector, minimize the number of unwanted directly penetrating gammas reaching the detector, and to maximize the detection efficiency in the area of interest.

Collimation schemes can take on an increasingly complex form. In studying the possible application of these forms to FFV we find that size and cost constraints require that we adopt the simplest collimation that will provide

reasonable results. Of the available collimation scheme the most economical is the single aperture detector shown in figure 5. In reviewing the physics of this type of collimator we find that through its construction, gamma flux that originates from outside of a solid angle defined by the aperture size and length is suppressed. We also find that since the collimator restricts the detector view to a solid angle the area of view is related to the source to detector distance. In applying a single aperture detector to the crated fresh fuel measurement environment, we find the solid angle field of view offered by the collimator will provide a means of determining the localized surface activity of the fuel bundle in question. If we now wish to determine the area defined by the collimator we find that the distance from source to detector plays as an important role as the collimator dimensions. This situation is not satisfactory since the detector to source distance is not a known factor in all FFV scenarios. As an example consider one possible case where the fuel bundle to be viewed is missing. Since the first bundle is missing the next available bundle (which in this case is the bundle immediately below) will be viewed instead. This second bundle, located in the next storage plane, is now further away from the detector, resulting in a larger solid angle defined area than would have been defined if the bundle had been in the first plane. By referring to figure 5 we find that the new area is linearly related to the square of the detector to object distance. As a result, we find that as the detector to object distance increases, an ever increasing area of gamma emitters are brought into view. Quantification of this effect shows that this effect nullified the inverse square law that governs the measureable flux at the detector site. The result is a measured flux which is invariant as the distance from the source to detector is varied, provided the source being viewed is sufficiently large so as



INTEGRATING TO FIND TOTAL FLUX GIVES

$$I(L) = \frac{S}{2} \int_0^R \frac{2r dz}{r^2} = \frac{S}{4} \ln \left( 1 + \frac{R^2}{L^2} \right) \quad \text{eq 1}$$

in the case of the collimator,  $R$  is a function of  $L$

ie  $R = kL$

therefore eq.1 reduces to

$$I(L) = \frac{S}{4} \ln \left( 1 + \frac{k^2 L^2}{L^2} \right) = \frac{S}{4} \ln (1 + k^2) = \text{constant}$$

as a result the flux observed by a simple collimator is invariant with distance, provided the area being observed is larger than the view area

FIGURE 5. SINGLE APERTURE COLLIMATOR

to completely fill the field of view of the detector. In practical terms, this means that a collimation system, under full field of view conditions (ie. small solid angle and large source) will not be able to distinguish between fuel bundles located in the top, mid or bottom planes, and will therefore be unable to detect if either of the bundles has been removed from the crate.

In the preceeding discussion, mention was made of full field of view criterion. If this condition is not met, the distance independence effect no longer holds and the simple collimation techniques again becomes workable. In order to minimize the distance independence effect certain rules must be imposed on the design of a collimator. Investigation of the problem indicates that as a result imposing these rules there are several possible design criteria, each of which results in a different collimator design. Of the various choices available, there exist only two which seem to be acceptable for use in the FFV environment. For the time being these will be labeled the simple and the complex collimator.

We start the investigation by studying the more complex collimation system shown in figure 6. This diagram shows a converging collimator which restricts its field of view to a converging solid angle. By adopting this type of collimator design we find that the field of view is maximized for top plane bundles while being minimized for second and third plane bundles. This design therefore completely eliminates the view area versus distance problem that is associated with the single aperture collimator design. In designing the converging collimator, the primary consideration is the convergence angle. In selecting this angle one must be aware of the tradeoff between optimization of the convergence effect and increasing difficulty in manufacturing and handling the actual collimator. When these tradeoffs are considered we find that the

handling and fabrication concerns are the constraining factors. In reviewing the requirements of the converging collimator we find that optimum situation would be one in which the vertex of the convergence cone would appear at or below the bottom fuel bundle. From the practical side we also find that the collimator must be shallow enough to fit in the space between each crate for on the shelf measurements or between the crate and floor for measurements made on crates removed from the storage shelf. In either case the maximum allowable thickness would be of the order of 3 inches. By using these two constraints, a computer model of the complex collimator was developed. The important parameters relating to the model are given in figure 7.

The second collimation scheme to be investigated was the simple collimator. This collimation scheme is identical to the single aperture collimator described earlier in this report, with the exception that the "full field of view" effect is minimized. This minimization is accomplished by ensuring that the solid angle of detection is sufficient so as to cover an area larger than that covered by a bundle which is not in the immediate plane of interest. The larger this angle, the closer the distance dependence will come to the theoretical maximum of  $1/(r^2)$ .

The theoretical basis for collimator design is quite simple. If one refers back to figure 5 and assumes that the actual photon producing body size is equivalent to area A1, the photon flux reaching the detector from such a source located at distances corresponding to the positions of A2 and A3, the detected flux will decrease. As the source size is decreased or the solid view angle is increased this effect will approach that which would be observed had the emitter (A1) been a point source.

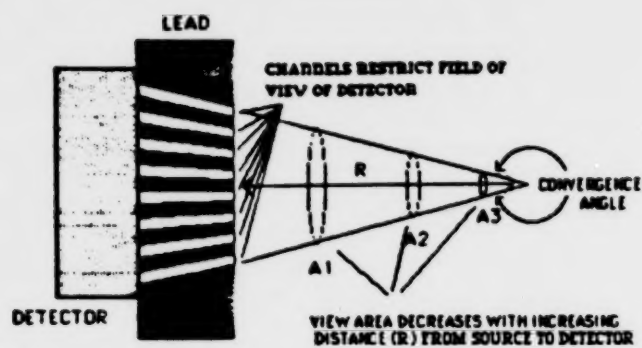


FIGURE 6 CONVERGING COLLIMATOR

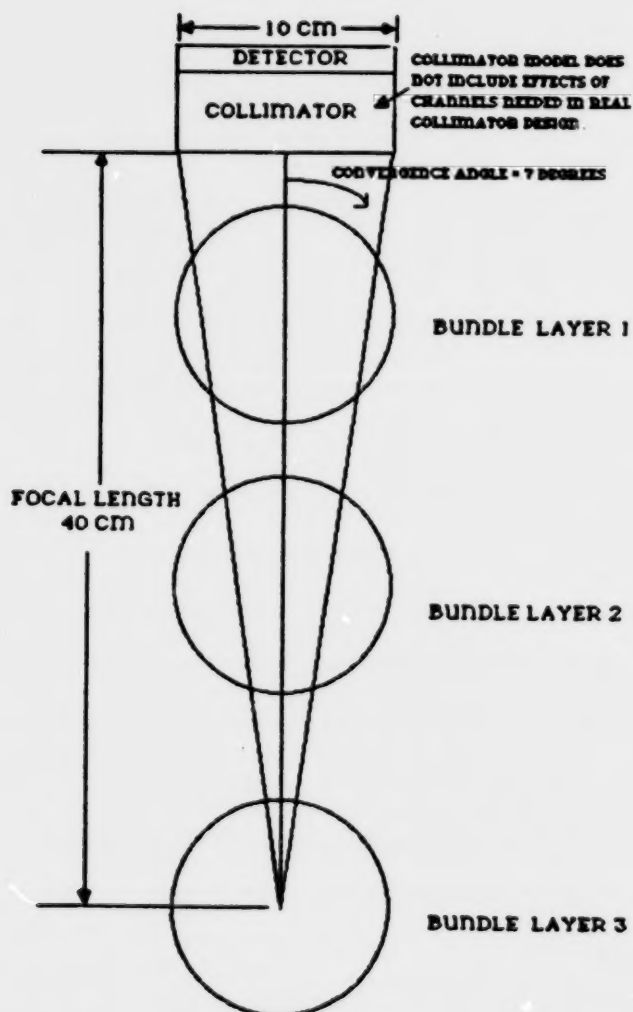
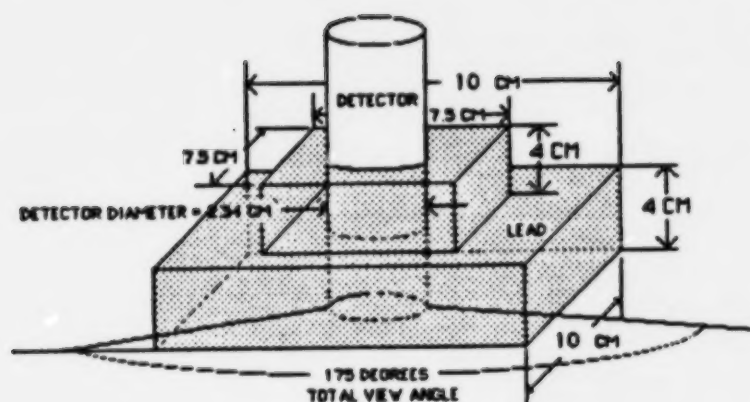


FIGURE 7. COMPLEX COLLIMATOR MODEL PARAMETERS

From examination of the theoretical aspects of the collimator and known fuel crate parameters we immediately find that we can expect somewhat less than ideal results from application of this technique. This conclusion results from the fact that the current fuel crate packing density is fairly high. As a result we find that even though the collimator view angle can be made large enough to make the view area of a fuel bundle negligible compared to the total view area, the side by side packing arrangement of the fuel bundles creates an essentially solid plane of fuel. This plane is so large as to preclude the possibility of increasing the view angle to an even larger value.

In light of these initial observations a simple collimator was devised. The resultant design is shown in figure 8. Two aspects of this design should be noted. The first aspect relates to the fact that the view angle is extremely large (175 degrees). This large angle was selected in order to provide maximum distance dependance. The second aspect of importance is that the actual detector is positioned in such a way so as to allow the detector to contact the fuel crate, therefore optimizing counting efficiency. The resulting design was tested via on site test and computer simulation.

Referring back to the initial discussion of gamma flux mapping the problem of fuel bundle shielding was described. If it is assumed that it is unsatisfactory to be unable to make a determination as to the presence of the center plane fuel bundles some means of mapping the internal flux of the fuel crate is required. Examination of the cross-section data for the fuel bundles indicates that it is certain that the fuel bundles in the top plane completely shield the bundles in the center plane. Because of this problem two additional collimation schemes were devised for use in making flux measurements from inside of the fuel crate.



MODEL ASSUMES NO SCATTERING IN COLLIMATOR MATERIAL

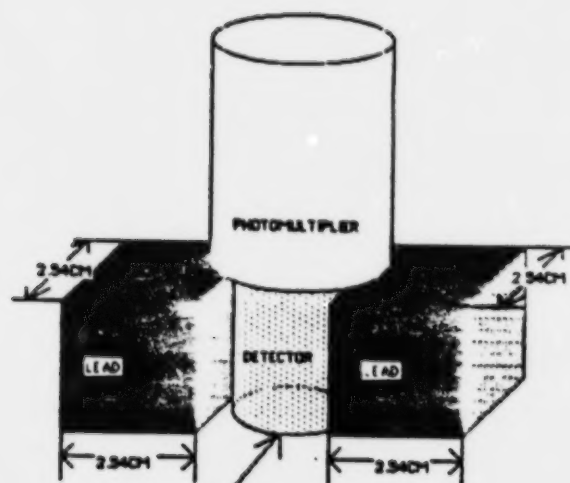
FIGURE B. SIMPLE COLLIMATOR DESIGN AND MODELING PARAMETERS

Both schemes are derivatives of the concept of the wide angle aperture used in the exterior scanning technique discussed above. Figure 9 shows both collimator designs. The actual technique to be employed for making these measurements involves placing the detector in the space between 4 fuel bundles, two of which are on each fuel plane. The detector would then be moved along the entire crate length with measurements being made at several points along the bundle length.

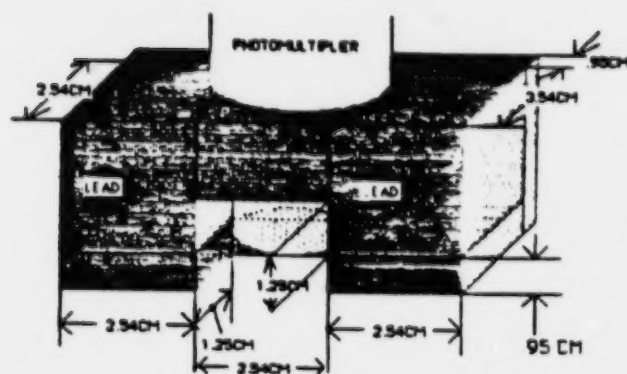
The first of the two internal collimators was designed to provide the detector with a full 360 degree (ungated), view of the surrounding bundles. This allows the detector to characterize a group of four bundles simultaneously. The second collimator provides a restricted (gated), view of one bundle. This would provide the ability to characterize a single fuel bundle. Each of these collimator designs has benefits and drawbacks. To illustrate these differences, table 5.1.1 has been prepared.

#### 5.1.2 CONDITIONS SIMULATED

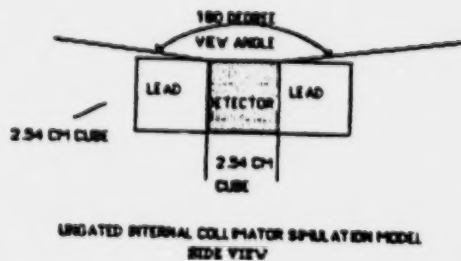
An attempt was made to simulate all possible situations in which fuel bundles were missing. To limit the computation time it was decided that the the number of missing bundles should be restricted to one. This decision was thought to be justified since a single missing bundle would represent the worst case situation that would have to be detected. Table 5.1.2 describes the various missing bundle geometries that were used during the simulation.



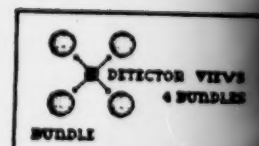
2.54 CM DIA  
360 DEGREE INTERNAL COLLIMATOR DESIGN



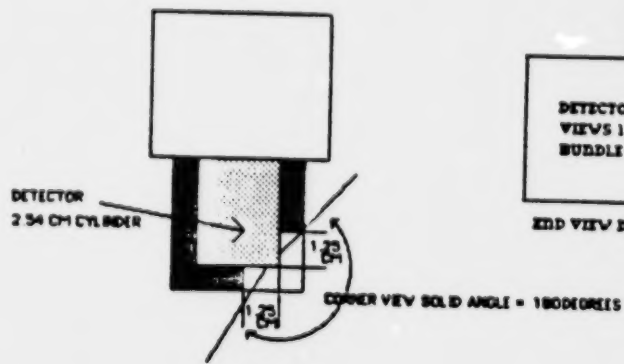
GATED INTERNAL COLLIMATOR DESIGN  
NOTE: DETECTOR OF SAME DIMENSIONS AS UNGATED DETECTOR



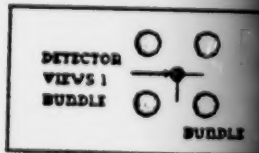
UNGATED INTERNAL COLLIMATOR SIMULATION MODEL  
SIDE VIEW



END VIEW DURING APPLICATION



SIDE CROSS SECTION OF INTERNAL GATED COLLIMATOR



END VIEW DURING APPLICATION

FIGURE 9. INTERNAL COLLIMATOR DESIGN USED DURING THE STUDY

Table 5.1.1 - Comparison of Various Collimator Designs

| Collimator Type             | Number of bundles<br>observed at one time | Benefits and<br>disadvantages                                                                                                                                                                                                                                    |
|-----------------------------|-------------------------------------------|------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| External                    | 1                                         | Does not see interior bundles but can be applied from outside of crate.                                                                                                                                                                                          |
| Ungated interior collimator | 4                                         | Can see mid plane bundles but cannot determine specific location of missing bundle. Must be used on interior of fuel crate since no background shielding is provided in backwards and side directions. Requires fewer measurements than the other two detectors. |
| Gated interior collimator   | 1                                         | Can see mid plane bundles. Can determine which specific bundle is missing. Awkward to use on exterior of fuel crate due to round shape. Requires many measurements.                                                                                              |

Table 5.1.2 - Missing Bundle Locations

| Test # | Bundle Location<br>Plane - Number |   | Measurement Vectors<br>(Refer to figure 4) |
|--------|-----------------------------------|---|--------------------------------------------|
| 1      | A                                 | 1 | A1-B1                                      |
| 2      | A                                 | 2 | A1-B2                                      |
| 3      | A                                 | 3 | A2-B3                                      |
| 4      | B                                 | 1 | A1-B1                                      |
| 5      | B                                 | 2 | A1-B2                                      |
| 6      | B                                 | 3 | A2-B3                                      |

## 5.2 SIMULATION RESULTS

### 5.2.1 FULL CRATE

As a test of CPGF, a complete spatial flux map was calculated for a full fuel crate. The procedure used in this test consisted of computing the photon flux along all of the vectors displayed in figure 4, while using the three simple (interior and exterior) collimators described previously. A run in which the more complex collimation scheme was also executed for a single vector so as to provide data in regards to the performance increase that could be achieved using the the more advanced collimation scheme. From figure 4 it will be noted that only top and middle plane vectors were used in the simulation. This was done as the bottom and top planes are symmetrically oriented around the middle plane. The computed results are plotted in figure 10. If one examines these plots, it will be noticed that the positions of the fuel bundles can be clearly identified as there is a significant elevation in the flux in the their immediate vicinity. It should also be noted that there are distinct differences between the calculated flux for the three collimator types. This difference is due to the fact that the different collimator designs allowed differing amounts of flux to reach the detector. The largest difference was noted between the gated and ungated internal collimator types. This is as expected as the ungated internal collimator allows roughly 4 times as much flux to reach the detector as does the gated version.

Referring to the complex collimator simulation results shown in figure 11 we find that the collimator clearly defines the bundle edge. Also it will be noticed that complex scheme provides a better background to peak ratio but a

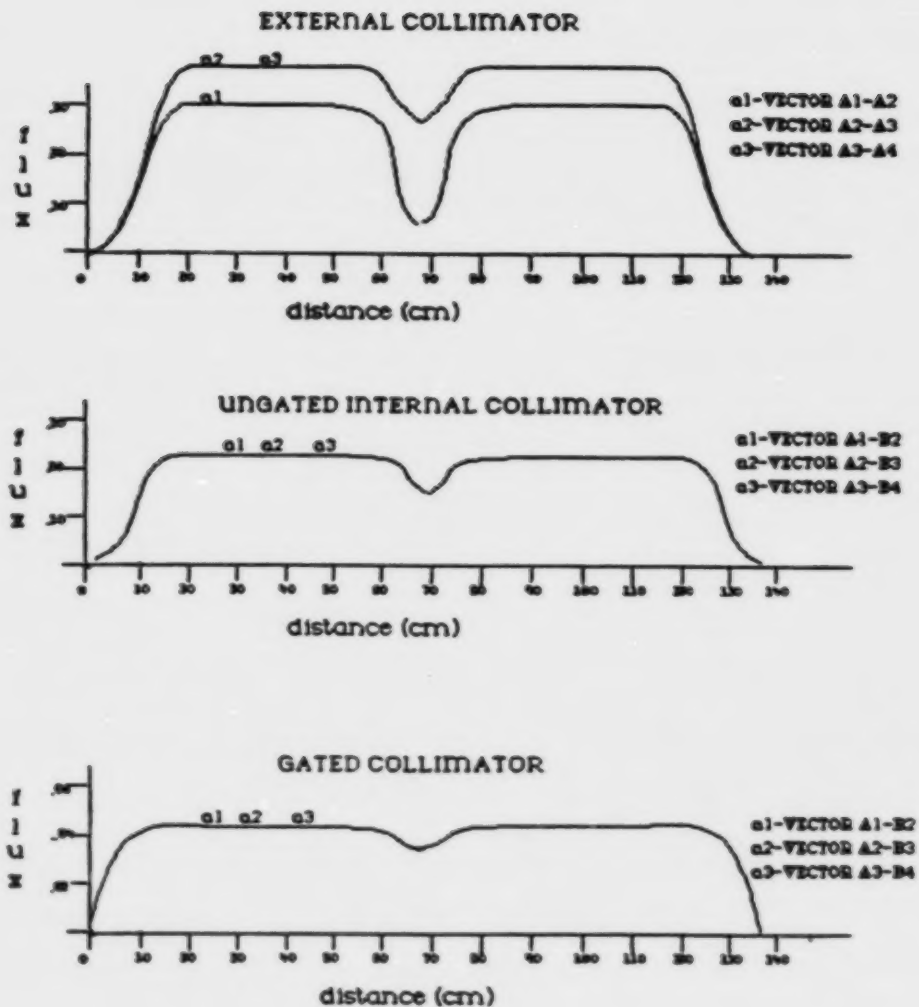


Figure 10. Full fuel crate flux calculations

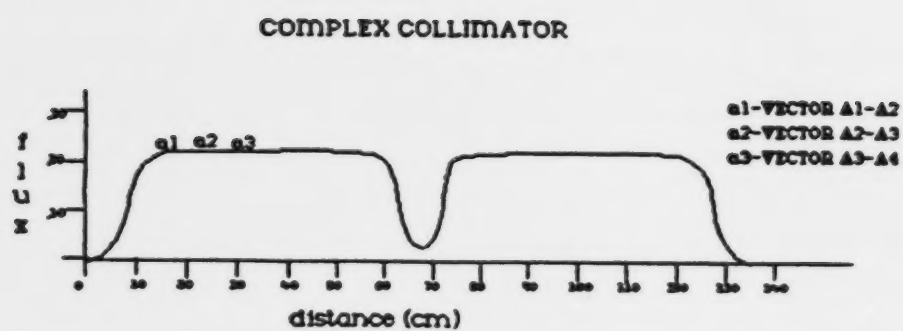


Figure 11. Full fuel crate complex collimator flux calculations

poorer absolute sensitivity than the simple collimation scheme. Since these results did not significantly vary from those obtained from the simple collimation scheme, it was decided that further study of the complex collimation scheme would not be warranted.

#### 5.2.2 MISSING TOP PLANE BUNDLE

Three geometries were run with fuel bundles missing from the top plane of the fuel crate. The procedure used during the simulation was to remove one of the top bundles and map the quadrant of the crate that contained the missing bundle. The calculations were carried out along the same vectors as shown in figure 4. Both the simple internal and external collimators were used for these tests. The resultant flux profile was then analyzed for deviations that could be used as an indicator of missing bundles. Only three geometries were run as it was assumed that no other missing single bundle locations were possible due to the symmetrical nature of the fuel packing.

Numerical results obtained from this series of tests are given in table 5.2.3 while the resultant data is plotted in figure 12. The numerical results represent the mid bundle flux values. In analyzing these results, it was felt that the performance of both the internal and external collimation schemes are quite acceptable. Of particular interest was the fact that the interior flux measurement was able to detect if a bundle was missing from the top plane. This supports the argument that the external collimators can be dispensed with entirely.

To summarize, we find that spatial gamma mapping can be achieved with a variety of collimation schemes and that each scheme can result in a differing missing bundle sensitivity.

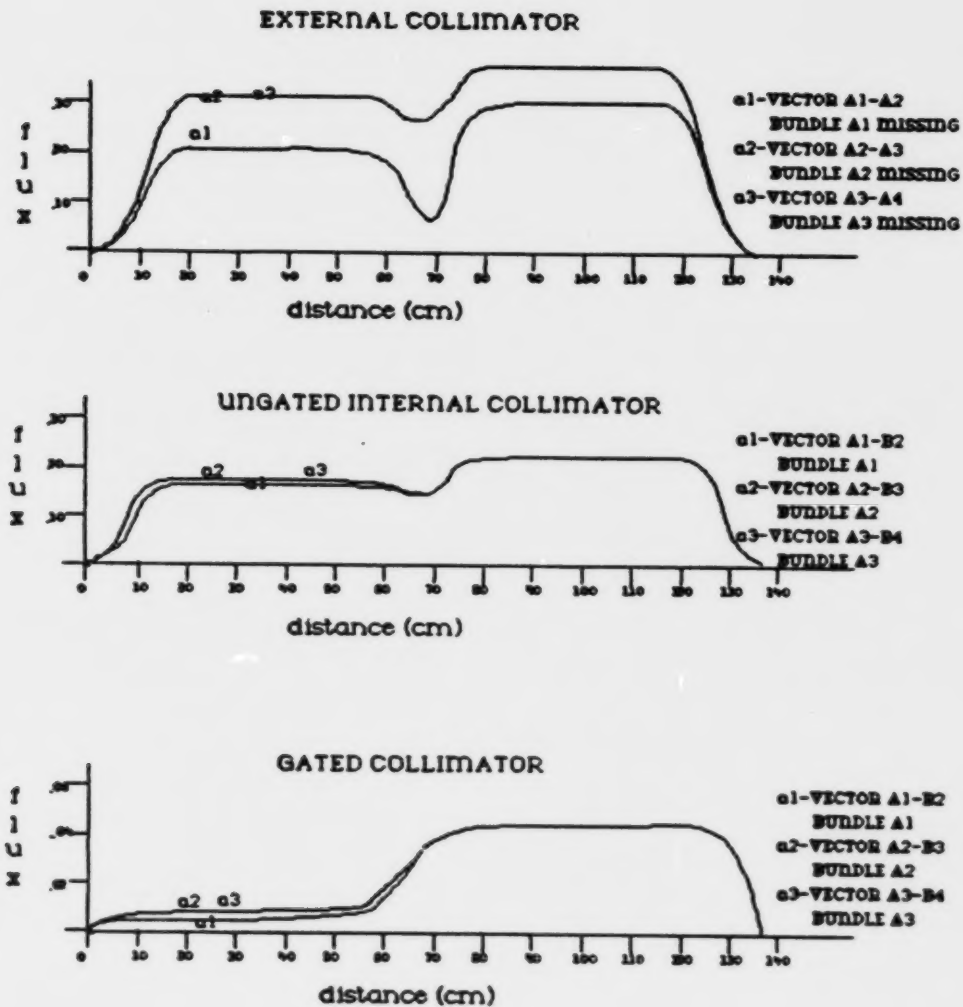


Figure 12. Top plane missing Fuel Bundle Calculations

### 5.2.3 MISSING CENTER PLANE BUNDLE

In order to determine the performance of the various collimation schemes where center plane fuel bundles are involved, a second set of simulations were undertaken. In this case, flux calculations were again carried out along the exterior and interior vectors shown in figure 4.

As expected, the results obtained from these simulations indicate that the exterior flux measurement will provide unsatisfactory results when internal bundles are to be characterized. Also, as expected, the results obtained from the internal flux calculations were found to same trends as those obtained with a fuel bundle missing from the top fuel plane. Figure 13 provides a record of the flux values obtained from the various simulations.

Table 5.2.3 documents the mid bundle worst case results obtained from the top and middle plane missing fuel bundle geometries. In viewing these results it will become obvious that measurement of exterior flux is of little use in the detection of fuel missing from the center plane. A difference will also be noted in the two types of internal collimation schemes. These differences can be attributed to the fact that ungated detector views a total of four bundles at one time, with the result being that the removal of one bundle can at most produce a 25% reduction in flux reaching the detector.

To summarize the findings, we find that the mid plane simulation set has shown that by using the interior collimation method, a positive identification may be made as to the presence of fuel bundles located in either the top, or center fuel crate planes. Through symmetry this also implies that the bottom plane is also observable. It has also been found that surface mapping is unable to detect fuel bundles missing from the center plane. Because of this

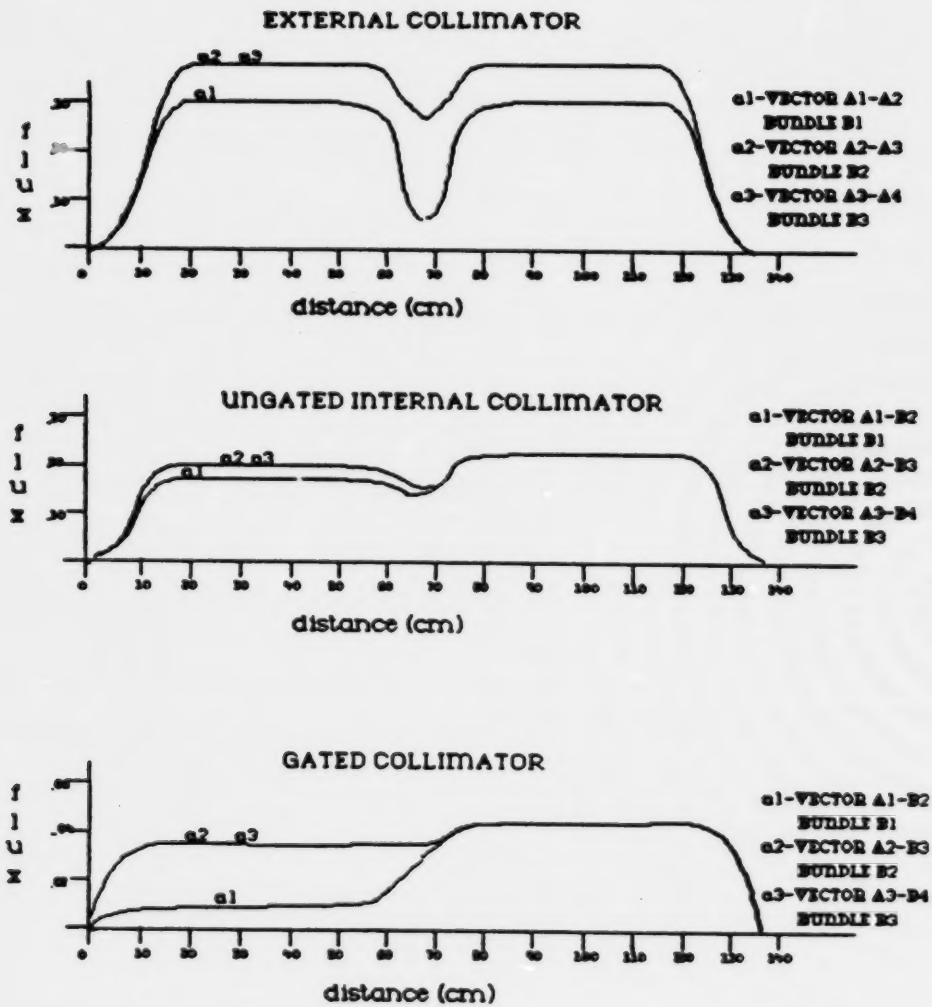


Figure 13. MID PLANE MISSING FUEL BUNDLE

limited usefulness, it is felt that further investigation of the surface mapping technique is not warranted.

Table 5.2.3 - Summary of Calculated Flux Values For Various Collimators And Missing Bundle Locations

| Test # | Missing Bundle Location | EXTERNAL |        |        | INTERNAL |         |        |
|--------|-------------------------|----------|--------|--------|----------|---------|--------|
|        |                         | Complex  | Simple | Vector | Gated    | Ungated | Vector |
| 1      | Top plane A1            | .40      | .72    | A1     | .10      | .75     | A1-B2  |
| 2      | Top plane A2            | .40      | .72    | A2     | .21      | .77     | A1-B2  |
| 3      | Top plane A3            | .40      | .72    | A3     | .21      | .77     | A2-B3  |
| 4      | Mid plane B1            | 1.00     | 1.00   | A1     | .21      | .77     | A1-B2  |
| 5      | Mid plane B2            | 1.00     | 1.00   | A2     | .78      | .90     | A1-B2  |
| 6      | Mid plane B3            | 1.00     | 1.00   | A3     | .78      | .90     | A2-B3  |

All values are relative to the full fuel crate values which are taken as 1.0 for each detector.

Table 5.2.4 - Mid Bundle Partial Bundle Flux Values for Uncollimated Detector

#### Orientation

Worst Case Normalized Flux Value  
Calculated for Uncollimated Detector  
Located 5.08 cm From Bundle Outline  
(relative to fuel bundle present value of 1.00)

|                                |      |
|--------------------------------|------|
| Missing half opposite detector | 1.00 |
| Missing half facing down       | .97  |
| Missing half facing detector   | .91  |
| Missing half facing up         | .97  |

#### 5.2.4 PARTIALLY MISSING FUEL BUNDLE

Several cases were run with partial fuel bundles. The partial missing bundle configuration was selected via BPGF so as to give a noticable change in the close range flux profile. Geometries that did not produce the desired close range flux change were those that saw fuel pencils removed only from the inner rows of the fuel bundle. This effect is attributed to the high shielding effect of the UO<sub>2</sub> pencils which results in the inner pencils being completely blocked from producing measureable fields. In the case where outer pencils were removed and the inner pencils left intact the resultant flux differed only slightly from that produced by a full bundle. As a result the only bundle configuration that would produce a detectable flux deviation was in the case of a half bundle where one half of the pencils were removed from the side of the bundle closest to the detector. This configuration was subsequently applied to the crate of fuel. During these runs the bundle orientation was set so as to simulate all possible rotations. These positions and resulting normalized flux values are shown in Table 5.2.4.

When these findings were applied to the center plane tests, the results varied, depending on which type of collimator was used. In the case of the gated collimators, we find that the partial missing bundle deviation was 8% in the best case. This detectability figure is low enough so as to make the signal difficult to separate from statistical fluctuations. Due to this low detectability, no results were tabulated.

In summary, it can be stated, that the gamma technique provides unsatisfactory results when used to detect the presence of partial fuel bundles. Because of these conclusions, it was felt that in field tests with partial fuel

bundles would not produce useful results and therefore should not be attempted.

One other case was investigated where lead (a similar weight material) is substituted for a missing fuel bundle. While no computer simulation was carried out for this test it is expected that the lead substitution will appear as a very large deficiency in flux. This is primarily due to the fact that lead, being a gamma absorber will assist in the minimization of background flux in the vicinity of the detector. As a result it is felt that there is strong justification to state that the gamma method is able to detect the substitution of a bundle with non radioactive materials.

### 5.3 NOTES REGARDING ENRICHMENT ANALYSIS

As discussed in appendix 1, enrichment determination can be facilitated through the selective counting of gamma photons which are emitted by U-235. This can readily be accomplished through the use of a variety of different types of energy resolving detectors.

The validity of the simulations carried out thus far as they relate to the task of enrichment analysis is ensured by the fact that the gamma flux calculations are based on a single energy algorithm. As a result, the flux calculated at the detector site is the actual uncollided flux reaching the detector. Since this flux is uncollided, it still retains its primary energy and can therefore be characterized by spectroscopic means.

An experiment to study the ability to carry out enrichment determination was not carried out due to the NaI detectors inability to obtain sufficient resolution to isolated the 185.7 Kev Gamma emission from the U-235 background.

None the less it is expected that it will be possible to make some type of measurement of the enrichment figure by using a higher resolution detector. In such a case it is expected that the time required to obtain satisfactory statistics will be in the order of 40 minutes. This figure has been derived solely through examination of the relative activities of the two isotopes of Uranium and as such is subject to a fairly high uncertainty.

#### 5.4 CONCLUSIONS DERIVED THROUGH SIMULATIONS

The results obtained from simulation of the gamma based FFV techniques indicate that by using flux mapping, it is possible to determine if one or more fuel bundles are missing from the fuel crate, without having to open the crate. Of particular interest is the fact that the simulations have shown that mapping of the external surface flux is not required to detect bundles that are missing from the top or bottom fuel planes. It has been found that a single set of internal measurements can be used to detect whether bundles are missing in either the top, middle, or bottom planes. The number of measurements needed to characterize a particular fuel crate can vary depending on what possible bundle anomalies are to be detected. In the best case where the presence of one complete bundle is to be verified only one measurement per bundle or bundle set needs to be undertaken.

It has also been shown that the gamma mapping technique has two shortcomings. These shortcomings have been listed below:

1. Inability to detect partially missing fuel bundles.
2. Access must be gained to the space between top, middle, and bottom fuel storage planes.

In addition to the problems listed above, it has been found that there are several positive points related to gamma mapping. These are:

1. Enrichment analysis is possible
2. Complete testing of a fuel crate is possible with a minimum of 12 and a maximum of 36 flux measurement vectors. The two figures correspond to measurements being taken with a single ungated and single gated internal collimation scheme respectively.

One point of interest not mentioned thus far is that even though the external collimators studied were unable to detect mid plane bundles, the technique was applicable from outside of the fuel crate. This would allow the technique to be applied "on the shelf" with no modification to the fuel crates required. The internal technique requires that the collimator be placed into the center of the bundle planes. It is also possible that this type of measurement can be carried out while the fuel is stored on the shelf but modifications would have to be made to the fuel crates. These aspects are discussed in chapter 10 of this report.

Since the gamma flux mapping technique has been shown to be theoretically workable, in field testing was carried out to verify the predicted performance.

## 6.0 ON SITE GAMMA BASED ASSAY EXPERIMENTS

Once the gamma mapping method was tested through simulation, in field tests were carried out using the modelled methods. These tests were carried out at both the Mississauga AECL lab and at PNGS. Results obtained from these in field tests were then compared to the results obtained via simulation. The following subsections detail the procedures used and results obtained from each type of test.

### 6.1 EXPERIMENTAL SETUP

The equipment used for the on-site testing consisted of the following:

Prototype of BOT ENGINEERING LTD. MODEL 100 COMPUTING MULTICHANNEL ANALYZER Configured to 2048 channel mode

ORTEC 475 Shaping amplifier configured for x5 gain and 1 Micro second shaping time.

ORTEC High Voltage power Supply set to 1000 V positive bias

Harshaw 2.5cm X 2.5cm cylindrical NaI/PMT gamma detector

Figure 14 illustrates the equipment configuration used for the experiment.

The collimators used during the course of these experiments were similar to those described in the collimator design section of this report (section 5.1.1). Mapping of the exterior of the fuel crate was undertaken using the exterior collimator while the interior mapping was carried with the two internal collimators. The design of these collimators are shown in figure 9. In order to allow sufficient space to place the detector assembly between the bundles, foam inserts were recovered from empty fuel crates and cut so as to provide an access channel between bundles. The detector and collimator were

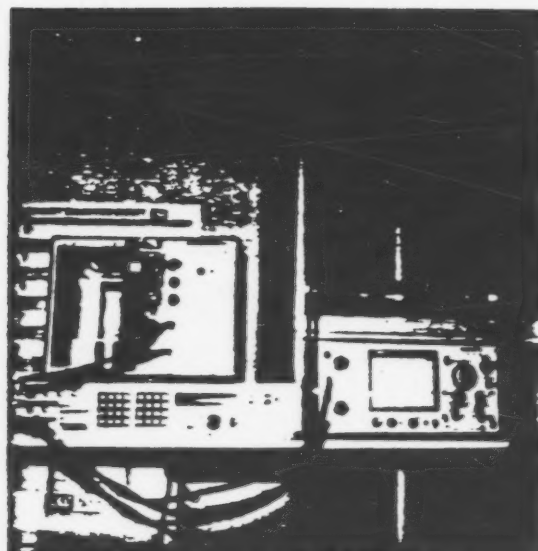
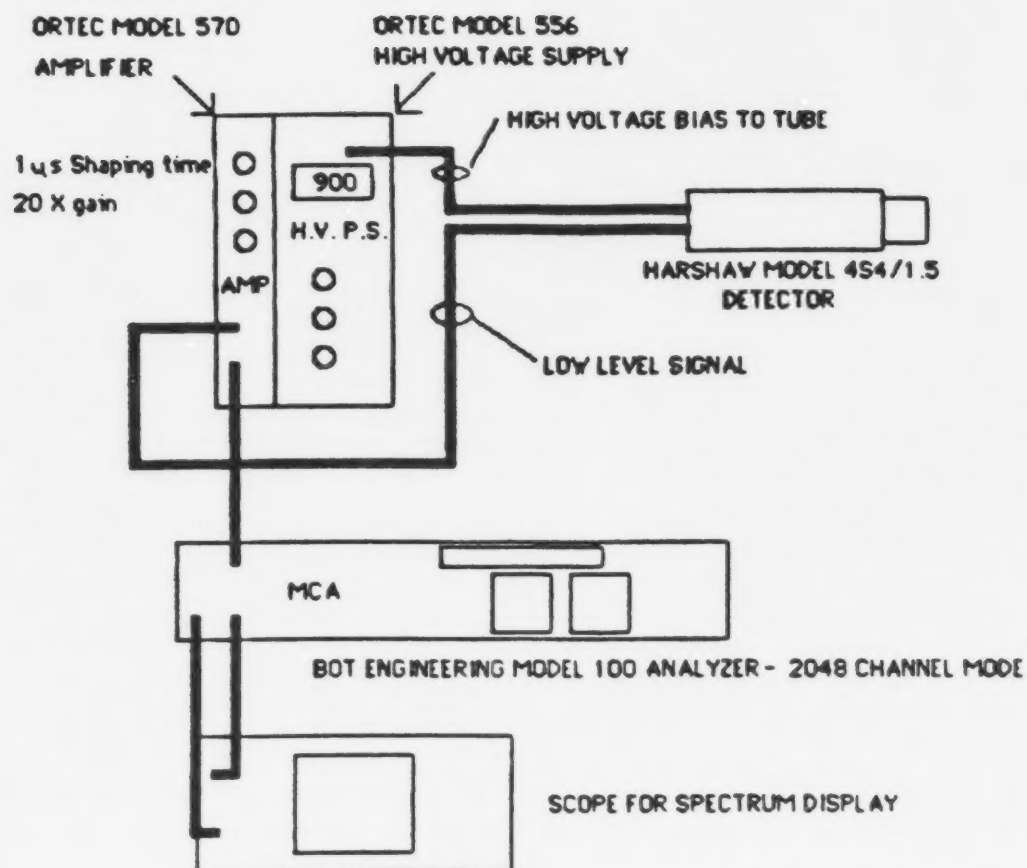


FIGURE 14. EXPERIMENTAL CONFIGURATION USED IN GAMMA TESTS

then moved along the newly formed channel and measurements taken at various points. Once the tests were complete the original undamaged inserts were returned to the test crate. The crates used for the experiments were of Westinghouse design and thereby offered equal bundle spacing across the entire fuel crate. This latter point is important because the General Electric design crate does not exhibit equidistance bundle spacing and thereby is not directly suitable for use in gamma based bundle measurements. This problem is discussed in the section of this report dealing with crate changes required for Gamma based FFV (section 10.1.2).

As the detector was moved from position to position, baseline compensated integrations of the energy band extending from 50 to 200 KeV were undertaken. This integral was then recorded as the primary flux value. The spatial density of measurement that was used during the testing procedure varied with the test being undertaken. In the test used for full crate simulation confirmation, the spatial measurement interval was 5.08 cm. For the missing bundle situation a measurement interval of 15.26 cm was used.

During the course of the tests, counting times of 60 seconds per point were used. This counting time resulted in an average integrated counts in the range of 700 to 1200. This provided a worst case statistical error of  $\pm 4\%$ .

#### 6.1.1 FULL CRATE

A full crate of fuel was used to confirm the proper operation of the simulation code and to obtain a baseline count rate from which missing bundles could be identified. The baseline count rates were measured using a full crate of fuel. In this case, baseline refers to reference count rate to which the

data collected from an unknown fuel crate could be compared. In an actual verification environment a discrepancy between the collected and baseline data would indicate a bundle deficiency. In order to derive a baseline for each collimation scheme a full fuel crate was mapped with the collimation schemes described in section 5.

The results obtained from the in field flux measurements are in shown in figure 15. For the remainder of this report these values have been normalized to 1.00 so as to allow simple comparison with the simulation results.

#### 6.1.2 MISSING TOP PLANE BUNDLE

The next in field test to be carried out consisted of the internal and external mapping of a fuel crate with one missing bundle. The measurement vectors used for this test were identical to those used for the missing top plane fuel bundle simulation. These are again as shown in figure 4 of this report. A spatial measurement interval of 15.24 cm was used for these tests. During the course of this test, only one fuel bundle at a time was removed from the test crate.

The results obtained from this test clearly show that there is a bundle deficiency. When the results are compared to those obtained via simulation, it is again found that the experimental data suffers from poorer resolution, and statistical noise than predicted. The magnitude of these anomalies are comparable to those obtained during the full crate test.

The results of these tests are shown in figure 16. From this figure, it will be noted that internal mapping was able to detect the bundle which was missing from the top fuel plane. This finding confirms the fact that external

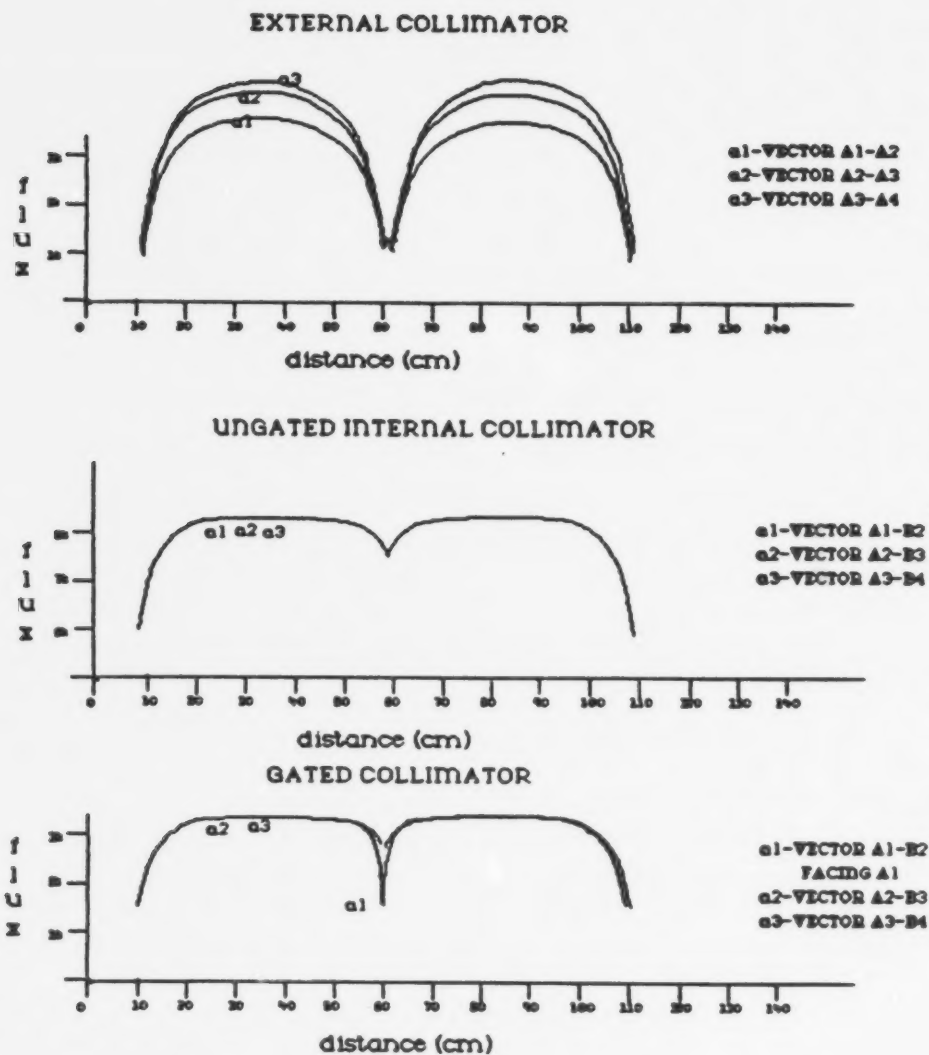


Figure 16. FULL CRATE GAMMA MEASUREMENT  
BEST FIT PROFILE

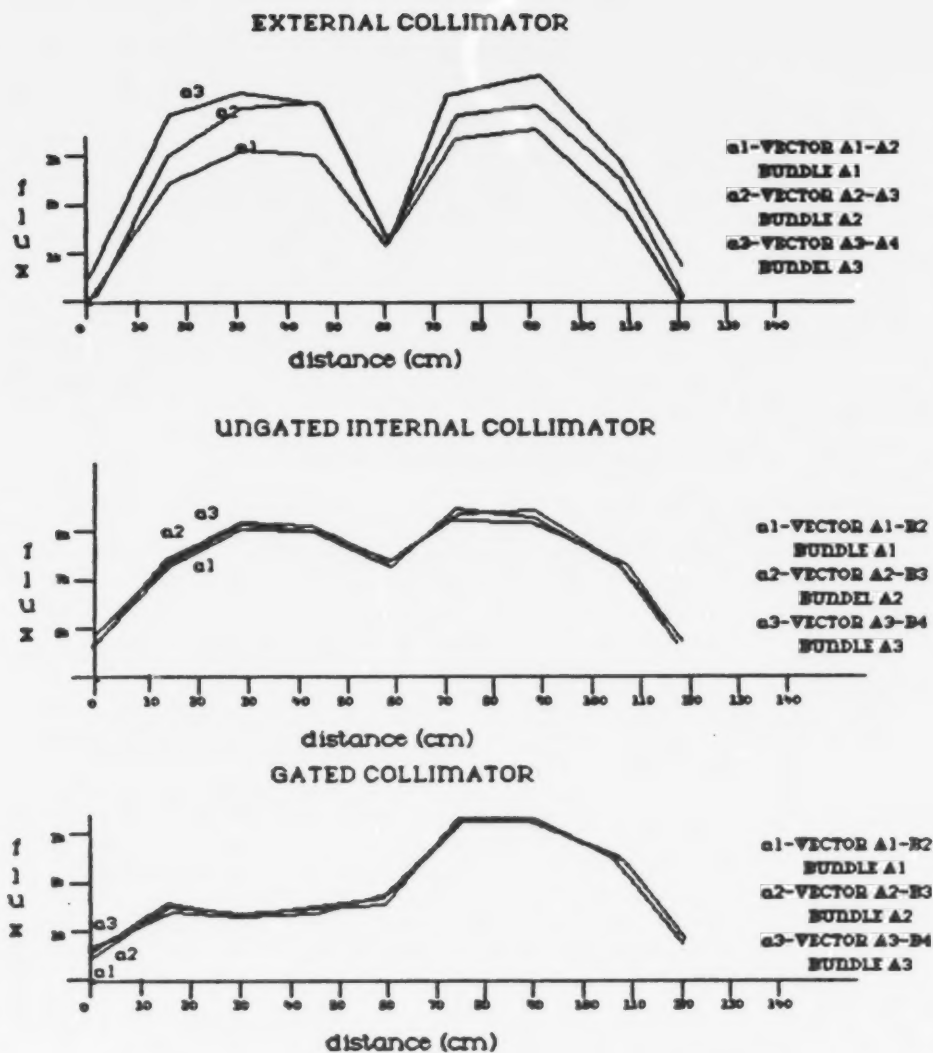


Figure 16. MISSING TOP BUNDLE GAMMA MEASUREMENT  
BEST FIT PROFILE

mapping is not required to detect fuel bundles that are missing from the top or bottom fuel planes. This result justifies the decision to totally eliminate the external mapping technique. Table 6.1.3 summarizes the results obtained from this test.

To summarize the findings associated with this test we have:

1. The same degraded spatial resolution and noise that was observed in the full crate case appeared in the missing bundle case.
2. The missing bundle location could be clearly identified with both the surface map and the internal map as an aberration in the flux profile.
3. The magnitude of the flux variation observed at the missing bundle location was up to 60% for the gated collimator, and up to 20% for the ungated collimator.
4. The experimental results obtained generally follow those obtained via simulation.

#### 6.1.3 MISSING CENTER PLANE BUNDLE

A final test was carried out in which a center plane fuel bundle was removed and both internal and external maps undertaken. As in the case of the top plane missing fuel bundle, the various measurement vectors were sampled with an interval of 15.25 cm.

The results obtained from the internal measurements were very similar to those encountered during the missing top plane bundle test. The results

obtained from surface mapping indicated that the surface flux could not be used to detect the missing fuel bundle.

The experimental results obtained from this test are shown in figure 17. Tabulated results can be found in table 6.1.3.

To summarize the results obtained from this experiment, we have:

1. The same degraded spatial resolution and noise that was observed in the other experiments was observed in this experiment.
2. The missing bundle location could be clearly identified via a flux depression in the spatial flux plot. This depression is observable for both internal and external measurements, with external being unable to detect missing internal bundles.
3. The magnitude of the flux variation observed at the center plane missing bundle site was of the order of 0% for the surface measurement, and 23% for the internal measurement
4. External flux cannot be used to detect bundles missing from the center plane.
5. Experimental trends generally follow those predicted by simulation.

Table 6.1.3 summarizes the measured missing bundle sensitivities for each type of collimation scheme for a variety of missing bundle locations. The values expressed are relative to the normalized full crate measurements.

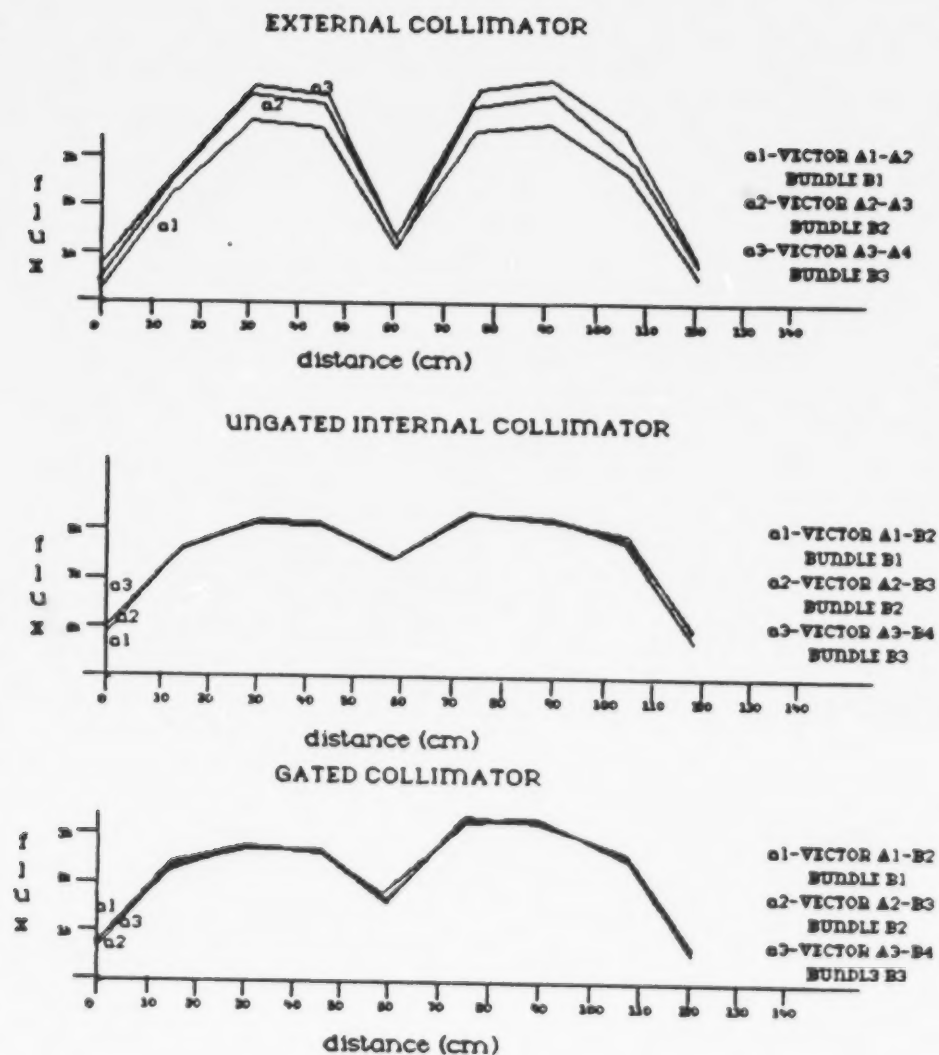


Figure 17. MISSING MID-PLANE BUNDLE GAMMA MEASUREMENT  
BEST FIT PROFILE

Table 6.1.3 - Normalized flux values measured at bundle center for one missing bundle. Values expressed relative to measurement taken at same location with no bundles missing. Errors expressed are cumulative, being sum from both full and missing bundle tests. Errors in actual measurements are 2%.

| Test | Missing Bundle Location | External Map (simple) | Vector | Internal Map (gated) | External Map (ungated) | Vector |
|------|-------------------------|-----------------------|--------|----------------------|------------------------|--------|
| 1    | Top plane A1            | .76 +\ - 4%           | A1     | .37 +\ - 7%          | .79 +\ - 5%            | A1-B2  |
| 2    | Top plane A2            | .79 +\ - 4%           | A2     | .42 +\ - 7%          | .84 +\ - 5%            | A1-B2  |
| 3    | Top plane A3            | .89 +\ - 4%           | A3     | .35 +\ - 7%          | .82 +\ - 5%            | A3-B4  |
| 4    | Mid plane B1            | .99 +\ - 4%           | A1     | .74 +\ - 5%          | .87 +\ - 4%            | A1-B2  |
| 5    | Mid plane B2            | 1.00 +\ - 4%          | A2     | .79 +\ - 5%          | .90 +\ - 4%            | A1-B2  |
| 6    | Mid plane B3            | 1.01 +\ - 4%          | A3     | .77 +\ - 5%          | .87 +\ - 4%            | A3-B4  |

## 6.2 CONCLUSIONS REACHED THROUGH EXPERIMENTATION

Several conclusions can be reached as a result of analyzing the experimentally derived spatial mapping data. These are:

1. Surface measurement can only detect the presence of missing top or bottom plane bundles, whereas internal measurement can detect both missing center plane and either top or bottom plane fuel.
2. The data derived from the mapping of the fuel crate is directly applicable to a go/no-go assessment of fuel bundle integrity provided go/no-go count values are derived for the midpoint of each bundle position.

## 6.3 OVERALL CONCLUSIONS AND RECOMMENDATIONS AS THEY RELATE TO GAMMA BASED FRESH FUEL VERIFICATION

Through simulation and experimentation, it has been shown that the gamma based FFV technique can be used to detect missing fuel bundles. During the course of these tests, it has been shown that internal flux mapping is better suited to the detection of all forms of bundle deficiency. As a result of these observations, the concept of external scanning has been abandoned.

To summarize our findings, we have:

1. Gamma techniques cannot detect partial bundles.
2. Gamma techniques can detect whole missing fuel bundles.
3. Point wise data analysis at bundle midpoint can be used for the detection

of missing fuel bundles.

4. Counting times of the order of 60 seconds per point provide adequate statistical error to determine if a bundle is present or missing.
5. A minimum of 12 measurements are required to fully characterize a crate of fuel when a single ungated detector is used.
6. Counting times of the order of 12 minutes are required to fully characterize a fuel crate using an ungated detector.

In conclusion, it can be stated with relative confidence, that the gamma measurement technique can be used for fresh fuel verification.

## 7.0 NEUTRON BASED FRESH FUEL VERIFICATION

### 7.1 PRINCIPLE

While a full review of basic neutron mechanics is beyond the scope of this report, a section of the appendix has been devoted to this task. As a result, the reader is referred to that section if a tutorial on the subject of neutron physics is required.

### 7.2 SIMULATION

As discussed in the appendix section, a particle tracking code is needed to accurately model the behavior of a neutron undergoing scattering and absorption in regions where sharp boundaries exist. The geometry associated with the use of neutrons for fresh fuel verification falls under this category. In light of this requirement, a Monte Carlo particle tracking code was developed to allow the simulation and verification of the proposed neutron technique. The resulting code, (which will be referred to as NEUT), was derived from a particle tracking code originally developed for the modelling of neutron moderation and capture in hydrogenous materials. Verification of the original code has been undertaken through comparison with the industry standard MORSE code.

In order to optimize the execution speed of NEUT, a center of mass isotropic scattering model for neutron nucleus scattering was adopted. While the center of mass scheme is only usable in situations where elastic neutron-nucleus interactions are dominant, the physical properties of fresh fuel allow its use. Because of the great savings in computer time afforded by this non

standard computational technique it has been possible to undertake significantly more simulations than would be possible with MORSE.

The output provided by NEUT, has taken the form of detector compensated wide band detected flux predictions, spatially plotted neutron distributions, and spectral energy profiles at the detector location. By using various results, it has been possible to determine the optimal configuration for use as an FFV tool.

#### 7.2.1 SIMULATION GEOMETRIC CONFIGURATION

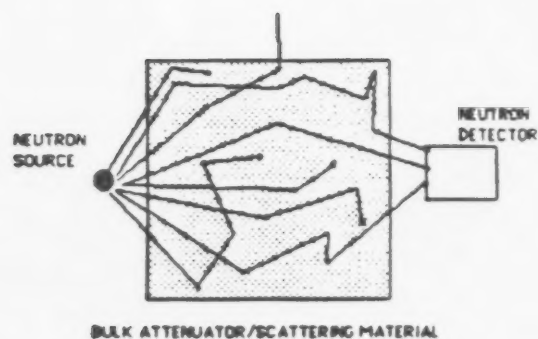
The geometric arrangement used during the simulation phase of the neutron based FFV development consisted of a modified classical transmission configuration. This configuration shown in figure 18 utilizes no shielding or collimation. Thus, unlike the gamma based FFV technique, collimators and shields are not required. This fortunate situation has been made possible by the fact that the areas where the FFV process is to take place are free from background neutrons. Also, initial simulations have also indicated that through proper configuration, it was possible to eliminate any crate to crate interference event when crates are side by side.

During the course of the simulations, a variety of source/detector/geometry configurations were investigated. These configurations included a variety of neutron source energies, detector energy response characteristics, and detector positions. In terms of the neutron source energy, the energies which corresponded to the peak emission of the most commonly available neutron sources were used. In particular, both the 2MeV and 6MeV energies were tested.

The detector characteristics were also tailored to simulate commonly available detectors. The response curves utilized during the simulations were the familiar  $1/v$  response common to He3 based detectors, the flat response which is characteristic of the so called black box detector, and a three group energy resolving detector (the  $1/v$  response refers to the inverse relationship between the neutron velocity and the detector detection efficiency). It should be noted that the non energy resolving, or group response detector (ie.  $1/v$  and flat energy response) are applied to the technique in an attempt to determine if the technique is applicable without need for energy spectrometry equipment.

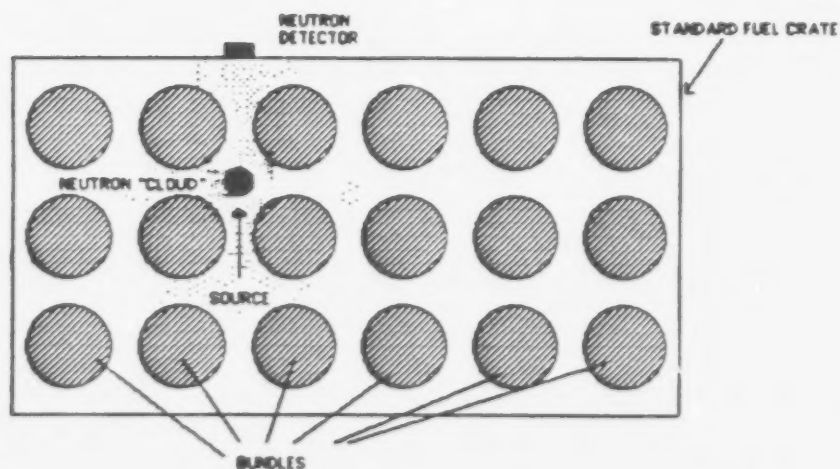
Two experimental geometries were adopted in the simulation phase of this study. The first configuration involved placing the detector and source on opposite sides the of the crate. The second, preferred configuration involved placing the source in the center of the crate and the detector on the exterior of the crate. This latter configuration is shown in figure 19. Note that this latter configuration requires that an access port be provided into the center of the fuel crate. While this requirement is a disadvantage from the point of having to change the crate design, it is felt that from practical terms it has advantages that warrant its use. One advantage is that placing the source in the center of the bundle array will result in a significant reduction in the strength of the source required to verify a crate. Also, the operators gamma dose received from the source will be greatly reduced through the shielding effect provided by the fuel bundles. This latter advantage would in turn eliminate the requirement that a heavy gamma shield be manipulated during the FFV procedure.

As in the case of the gamma simulations and experiments, symmetry was used where ever possible. This has resulted in simulations being carried out on



CLASSICAL NEUTRON ATTENUATION CONFIGURATION  
SHOWING SCATTERED AND ABSORBED NEUTRONS  
ANY CHANGE IN BULK PROPERTIES OF ATTENUATOR  
WILL RESULT IN CHANGE IN NEUTRON FLUX VARIATION  
AT DETECTOR LOCATION

FIGURE 18 CLASSICAL NEUTRON TRANSMISSION CONFIGURATION



NEUTRON SOURCE EMITS "CLOUD" OF FAST NEUTRONS  
SOME OF WHICH ARE ABSORBED AND/OR SCATTERED  
BY FUEL BUNDLES. REMOVAL OF ONE BUNDLE OR CHANGING  
THE PROPERTIES OF A BUNDLE RESULTS IN DIFFERENT  
NEUTRON FLUX REACHING DETECTOR  
THE RANGE OF THE NEUTRON CLOUD IS APPROXIMATELY  
1 BUNDLE DISTANCE IN THE CASE OF Cf-252 (2 MeV)  
NEUTRONS AND 2 BUNDLE DISTANCES IN THE CASE OF  
Am-Be (6 MeV) NEUTRONS

FIGURE 19 CONFIGURATION USED FOR NEUTRON  
SIMULATION

"quadrants" of the fuel crate. The lines that delineate the measurement quadrants are shown in figure 20.

#### 7.2.2 CONDITIONS SIMULATED

Due to the large number of detector/source/geometry permutations that can be adopted for the neutron based technique, a limit on the number of variables had to be imposed. The result was that the investigation was limited to two source types and several parametrically determined detector configurations. The source types that were selected corresponded to the two most commonly available isotope sources, Am-Be and Cf-252. The detector simulated locations were chosen by running a test case where the source was placed within the fuel crate and the neutron trajectory tracked until it intercepted a simulated position sensitive detector surrounding the crate. The results obtained from this simulation were then compared to a case where the same type of neutron tracking was carried out with a fuel crate that was missing one bundle. By comparing the neutron-detector collision density in both cases and selecting the point of maximum difference, the optimum detector position was obtained. During the course of this investigation, it was found that the optimum detector location was simply the area where the detector was within two bundle dimensions of the missing bundle. Extrapolating from this finding, several detector locations were chosen so as to provide optimum detection efficiency while minimizing the number of detector locations needed to characterize a crate of fuel.

After the various simulations were completed, in field tests were undertaken as a confirmation. The in field tests are described in section 8.0

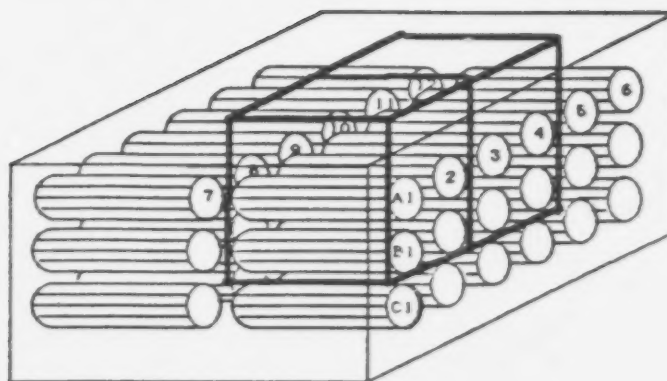
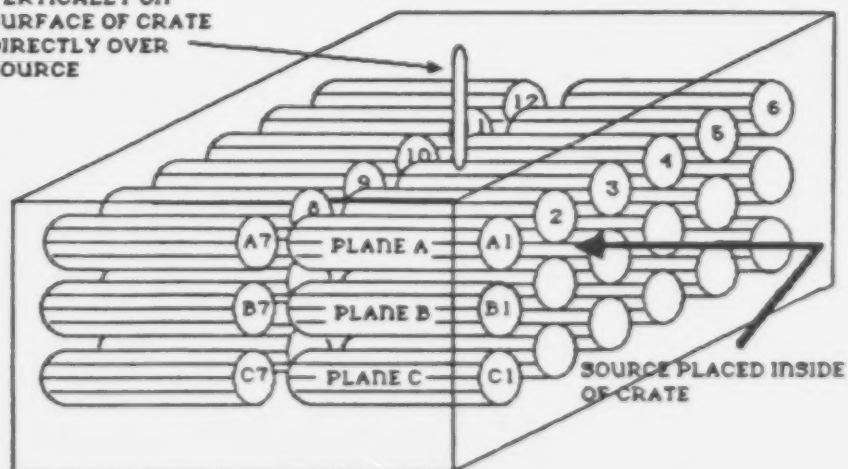


FIGURE 20 MAP OF QUADRANTS REFERED TO IN TEXT

DETECTOR PLACED  
VERTICALLY ON  
SURFACE OF CRATE  
DIRECTLY OVER  
SOURCE



IDENTIFICATION OF POSITION IN FUEL CRATE IS  
THROUGH PLANE AND BUNDLE NUMBER.

-FOR MID-PLANE OR MID-BUNDLE LOCATIONS THE  
LOCATION IS IDENTIFIED BY GIVING THE  
IDENTIFICATION OF THE BUNDLE THAT SURROUNDS  
THE POINT IN QUESTION

-MEASUREMENTS TAKE PLACE DIRECTLY ABOVE SOURCE  
ON SURFACE OF CRATE. DETECTOR IS PLACED PERPENDICULAR  
TO CRATE SURFACE

**POSITIONING EXAMPLES:**

- A1 - TOP PLANE BUNDLE 1
- B6 - MID PLANE BUNDLE 6
- A1-B1 - BETWEEN BUNDLE A1 AND B1
- C2-C3 - BETWEEN BUNDLE C2 AND C3
- B2-B3 - BETWEEN BUNDLE B2 AND B3
- IN CENTERLINE OF BUNDLE
- A1-B2 - MID PLANE IN THE CENTER OF  
BUNDLES A1, A2, B1, B2

FIGURE 21 - BUNDLE, SOURCE AND DETECTOR  
POSITION MAP

and use the same conventions adopted in this section.

### 7.3 SIMULATION RESULTS

The simulation results are discussed in the following sections. Each section describes simulations carried out for each missing bundle scenario. Finally, each section is further subdivided so as to delineate Cf-252 and Am-Be results. Due to the quantity of simulations carried out, the following text details only the qualitative aspects of the tests, with the quantitative results being documented in the appropriate tables.

#### 7.3.1 FULL CRATE

One simulation was run in which a full crate of fuel was irradiated by the neutron source. The results of this exercise were then used to determine the transmission efficiency, degree of thermalization and effective interaction distance, (lifetime), of the source neutrons. This data was then used to determine the minimum usable source strength that could be used for the in field test. Also, the full fuel crate data is extremely important as it serves as a baseline from which the results from missing bundle tests could be compared.

Since it was earlier determined that the source to detector distance must be no more than two bundle dimensions for optimal technique performance, the source and detector were repositioned several times to cover the entire crate. Even though all simulation results have been tabulated, only the worst case results are discussed in the sections that follow.

The response of the various detector models were monitored as the

detector and source positions where varied. From this study it was found that although a flat response detector could in some cases provide up to 3 times more sensitivity than that obtainable from the 1/v detector, most cases produced gains ranging from 1.5 to 2 times. By examination of the energy resolving detector model output it was also determined that energy resolution was of little use in the case of Cf-252 but could be of benefit in some cases where missing bundles are to be detected by using the 6 MeV source. Despite these potential gains it was decided that the practical considerations of low availability and high cost outweigh the limited gains obtained from the use of these detector types. As a result, all further study has been carried with the 1/v model which corresponds to the commonly available He-3 gas filled detector.

In the case of the Am-Be source, it was found that the detector would in the worst case detect  $.00004 \pm 7\%$  neutrons for each neutron released from the source. These results were obtained for source a source position of B1-B2 and a detector position located on the surface above bundle A1. This corresponds to the detector and source near the edge of the fuel crate. In the case of the Cf252 (2 MeV) source, we find that the worst case neutron economy is  $.00001 \pm 5\%$ . This worst case efficiency was obtained at the source and detector position as observed in the Am-Be case. These neutron economies are of interest as they, together with permissible counting time, dictate what the permissible source strength will be.

The data collected for each source and detector position is of interest in only relative terms, therefore the figures obtained for each position are taken as a normalized value of 1.00. All other simulation results are taken relative to these values. Note that there is one normalized value for each detector/source position combination, and that these must correspond to missing

bundle data taken at the identical detector/source position. Since all values are normalized to 1.00 the results obtained from these simulations have not been tabulated.

### 7.3.2 MISSING TOP PLANE BUNDLE

Unlike the gamma mapping technique which is based on mapping the positions of each bundle, the neutron technique is based on average relative flux measurement. As such, the neutron flux values obtained from various crate measurements give no specific data as to the presence of a specific fuel bundle. Because of this averaging or gross view effect, collected data is only meaningful when compared to data collected from a full crate with the detector and source located at the same position on the crate.

Since it was impossible to predict which source/detector position combinations would produce optimal detectability of missing fuel bundles, several combinations of source energy, source position and detector position were simulated.

For the case of missing top plane bundles, it was required that three simulation pairs be carried out, one for each possible missing bundle location. Earlier study of the neutron technique indicated that the technique would be usable only if the source/detector to missing bundle distance was no greater than the length taken up by two adjacent crated bundle locations. In order to test the validity of this conclusion as it applied to the missing bundle case, several simulations were carried out where the missing bundle position was greater than this limiting value.

The Cf-252 simulation under the missing top plane bundle scenario indicates that missing bundles are reliably detected only if the the bundles are examined in groups of four (one bundle effective length). To accomplish this, one must insert the neutron source into the center of a group of four bundles (much in the way that the gamma probe is inserted). Applying this requirement to the characterization of a full crate of fuel, we find that complete crate characterization requires that we take measurements at a minimum of 12 detector/source locations. This corresponds to three locations for the front top plane, three locations for the front bottom plane, three locations for the back top plane, and finally three locations in the back bottom plane. As mentioned earlier in this report a provision for placing the source into the center of the fuel crate would have to be provided. Also the detector would have to be placed on the exterior of the fuel crate for each measurement. The missing top plane fuel bundle simulations indicate that a worst case missing bundle deviation of  $23\% \pm 7\%$  can be expected. In other words, the observed count rate varies by at least 16% if the fuel crate is missing one complete bundle provided that that bundle is within one bundle position of the source and detector.

The results obtained from the Am-Be (6 MeV) simulation indicate that the effective range of the source neutrons are increased due to the high mean neutron energy. From this effect it appears that it is possible to fully characterize a fuel crate with a total of 6 source locations and 12 detector locations. This result would lend itself to dual channel counting. This reduction in total number of required source locations results from the fact that the Am-Be source can be positioned in the center plane, midway between two fuel bundles. Each of these source locations then can be used to detect missing

fuel bundles in either the top or bottom planes. In quantitative terms, we find that when the Am-Be source is located in the center plane of a fuel crate that is missing either a top or bottom plane bundle, a minimum of 20%  $\pm$  4% (bundle A1 source position B1-B2) deviation in detected neutron flux can be expected. Since the Am-Be source model lends itself to simultaneous dual channel collection and requires fewer source manipulations, it is considered as the superior source in terms of top and bottom plane investigations

To determine if this conclusion is also valid in the case of a bundle missing from the center plane, a set of simulations were carried out. These are described next.

### 7.3.3 MISSING CENTER PLANE BUNDLE

When the two source types are utilized in a situation where a center plane fuel bundle is removed, we find that the missing bundle detectability is comparable to that achieved for the top and bottom plane tests. In examining these results we find that the 6 MeV source is applied to the missing center plane bundle scenario, we find that the worst case deviation for a single missing fuel bundle is 13%  $\pm$  5%. The results obtained for the 2 MeV source indicate that a worst case difference should a fuel bundle be missing is 23%  $\pm$  6% .

From these observations it is expected that either the Cf-252 or Am-Be source will provide usable results if used for the detection of missing fuel bundles. As can be seen from the theoretical worst case deviations, there exists a trade off between the use of Cf-252 and Am-Be sources. The first

tradeoff relates to the ability to detect missing bundle difference, while the second relates to the time required to verify a complete fuel crate. Other practical tradeoffs exist in the selection of the best source type but discussion of these points will be deferred until the appropriate section.

#### 7.3.4 EXTERNAL SOURCE CONFIGURATION

A study was undertaken in order to determine if it would be possible to verify a fuel crate without gaining access to the interior of the crate. The result of this study indicated that while it was possible to detect neutrons being transmitted through the crate, the flux deviations observed in cases of missing fuel bundles was too small to be of use (best case of 6%). As a result, this option was abandoned at an early stage.

#### 7.3.5 ENRICHMENT STATUS AND SUBSTITUTE EMITTER DETERMINATION

As an integral part of the simulation study, the fission contribution to the total detector neutron flux was calculated. By monitoring this figure while changing the U235 content of the fuel bundles, the possibility of total fuel crate enrichment determination can be confirmed.

The results from this investigation indicated that neither the Cf-252 or Am-Be sources could induce the fuel to produce sufficient quantities of fission neutrons to make a statistically valid determination of enrichment by direct observation of the detected neutron count rate. This lack of detectability comes about as a result of the large statistical error associated with the low

efficiency of the technique. As an example, we find that during irradiation by Am-Be neutrons the natural fuel fast fission neutrons will increase the detected count rate by 1.5%. While significant in terms of fission rate, this value is insufficient to be separable from the statistical noise (or counting error) associated with the detection of the scattered source neutrons. If one wishes to proceed to a gated detection or pulsed neutron source, it is likely that it would be possible to measure enrichment status via the delayed neutron output of the fresh fuel crate. Since each of these techniques involve highly specialized and expensive equipment, no further investigation was carried out.

In order to determine the ability of the technique to detect the substitution of materials, several simulations were carried out with lead substituted for fresh fuel. Lead was selected as a substitute since its density approximates that of the UO<sub>2</sub> fuel.

The results obtained from this simulation indicate that the substitutions of lead for fuel can be detected by observing an increase in detected neutron count rate. Comparing this observation to those encountered in the missing bundle case, we find that material substitution results in an increased count rate, whereas removal of a bundle results in a decrease in count rate. The order of magnitude of these effects are +10% and 20% respectively. Extrapolating from these results we expect to find that other substitute materials, having nuclear properties different from those of natural fuel, will produce count rates that deviate from those observed with a full fuel crate.

### 7.3.6 TABULATED RESULTS

The findings detailed in sections 7.1 to 7.3.5 were based on data that have been tabulated Table 7.3.6. The convention used in the definition of the source, bundle and detector positions is shown in figure 21 and is identical to that used in the gamma study (figure 4). Arrows are used to indicate what results are of significance to the discussion in the text. A note as to the significance of the finding is given where applicable. Note that symmetry will hold for all findings and therefore the detectability of only two center missing bundles indicates detectability of all center bundles.

Table 7.3.6 - Missing Bundle Normalized Flux Simulations Results

| Source Position<br>(and type) | Bundle<br>Location | Relative<br>Flux | Detected | Detector<br>Position<br>(surface)<br>(of crate) | Significance                                                    |
|-------------------------------|--------------------|------------------|----------|-------------------------------------------------|-----------------------------------------------------------------|
| A1 - B2 (Cf-252)              | B1                 | .77              | Yes      | A1-A2 <                                         | Detect all<br>missing<br>bundles on<br>one side of<br>crate end |
|                               | A1                 | .74              | Yes      | A1-A2 <                                         |                                                                 |
|                               | C1                 | .87              | Yes      | A1-A2                                           |                                                                 |
|                               | A2                 | .78              | Yes      | A1-A2 <                                         |                                                                 |
|                               | B2                 | .77              | Yes      | A1-A2 <                                         |                                                                 |
|                               | C3                 | .99              | No       | A1-A2                                           |                                                                 |
| A2 - B3 (Cf-252)              | B2                 | .86              | Yes      | A2-A3                                           |                                                                 |
|                               | B1                 | .80              | Yes      | A2-A3                                           |                                                                 |
|                               | C2                 | .78              | Yes      | A2-A3                                           |                                                                 |
|                               | A2                 | .94              | No       | A2-A3                                           |                                                                 |
| A3 - B4 (Cf-252)              | B3                 | .46              | Yes      | A3-A4 <                                         | Detect all<br>missing<br>bundles in<br>center of<br>crate       |
|                               | B2                 | .86              | Yes      | A3-A4                                           |                                                                 |
|                               | A3                 | .71              | Yes      | A3-A4 <                                         |                                                                 |
|                               | C3                 | .99              | No       | A3-A4                                           |                                                                 |
|                               | A1                 | .88              | Yes      | A3-A4                                           |                                                                 |
| A3 - B4 (Am-Be)               | C3                 | .78              | Yes      | A3-A4 <                                         | Detect all<br>missing<br>bundles<br>in center<br>of crate       |
|                               | B3                 | .53              | Yes      | A3-A4 <                                         |                                                                 |
|                               | B2                 | .88              | Yes      | A3-A4 <                                         |                                                                 |
|                               | A3                 | .66              | Yes      | A3-A4 <                                         |                                                                 |
|                               | B1                 | .96              | No       | A3-A4                                           |                                                                 |
| A2 - B3 (Am-Be)               | B3                 | .63              | Yes      | A2-A3                                           |                                                                 |
|                               | B4                 | .89              | Yes      | A2-A3                                           |                                                                 |
|                               | A4                 | .97              | No       | A2-A3                                           |                                                                 |
|                               | A3                 | .75              | Yes      | A2-A3                                           |                                                                 |
|                               | B2                 | .90              | Yes      | A2-A3                                           |                                                                 |
|                               | A2                 | .82              | Yes      | A2-A3                                           |                                                                 |
|                               | B1                 | 1.15             | No       | A2-A3                                           |                                                                 |
| B1 - B2 (Am-Be)               | A1                 | .80              | Yes      | A1-A2 <                                         | top bundles<br>at end of<br>crate                               |
|                               | A2                 | .71              | Yes      | A1-A2 <                                         |                                                                 |
|                               | B1                 | .87              | Yes      | A1-A2 <                                         |                                                                 |
|                               | B2                 | .63              | Yes      | A1-A2 <                                         |                                                                 |
| B3 - B4 (Am-Be)               | A3                 | .70              | Yes      | A3-A4 <                                         | top bundles<br>in center<br>of crate                            |
|                               | B3                 | .56              | Yes      | A3-A4 <                                         |                                                                 |
|                               | B2                 | .77              | Yes      | A3-A4 <                                         |                                                                 |
|                               | A2                 | 1.11             | No       | A3-A4                                           |                                                                 |
|                               | B1                 | .98              | No       | A3-A4                                           |                                                                 |
|                               | A1                 | 1.09             | No       | A3-A4                                           |                                                                 |

#### 7.4 CONCLUSIONS DERIVED THROUGH SIMULATION

As a result of the simulations that have been described on this section, it has been determined that the neutron based fresh fuel verification technique is theoretically feasible. This statement holds true for both types of commonly available neutron sources (Am-Be and Cf-252). The simulations have also shown that if an Am-Be source is used in the technique that a complete crate of fuel can be characterized with a maximum of six pairs of simultaneous measurements (12 total) and 6 source manipulations.

The overall utilization factor of the neutron source has been found to be .00001 in a worst case situation (Cf-252), and the overall sensitivity of the technique has been found to be a minimum of 13% count rate deviation per missing bundle using Am-Be. The investigation into fuel constituents indicates that the most likely substitute materials will register as a positive increase in detected count rate. This effect ensures that should a fuel bundle be substituted with some other material, the substitution would be detected during crate verification. It has also been determined that direct measurement of fuel enrichment is not feasible, however it has been found that the number of fast fission events that occur in the unenriched fuel during stimulation with an Am-Be source may be sufficient to determine enrichment status via delayed neutron techniques. Due to the complexity and cost of the equipment required to undertake delayed neutron yield measurement, it should be concluded that while enrichment determination is theoretically possible via the use of fast neutrons it is not obtainable via simple means and therefore is a poor candidate for use as an in field FFV technique.

In reviewing the characteristics of the neutron based FFV technique, we

find several interesting points. These are:

1. The removal of a fuel bundle decreases the neutron count rate. The cause of this effect has been studied and it has been determined the fuel bundles operate as reflectors that direct the source neutrons from a trajectory that would initially miss the detector to a trajectory that collides with the detector. Removing a bundle will therefore reduce the number of source neutrons reaching the detector. It has been proven through simulation, that this relection process is highly specific, with differences in both absorption and scattering cross-sections affecting the result.
2. While it is possible to detect if a crate is missing a bundle, it is impossible to determine the exact bundle position. This result is similar to that obtained via the use of the ungated gamma imaging scheme.

In conclusion, the theoretical study carried out in this section indicates that the neutron technique is capable of providing a go/no-go assessment of the contents of a fresh fuel crate. This decision would be based purely on the neutron count rate observed at locations on the crate exterior as a neutron source is inserted into the fuel crate at defined locations. Calculations based on theoretical models indicate that the observation of a count rate that deviates by more than  $\pm 10\%$  from the normal full crate can be considered as indication that the fuel crate under test has been tampered with.

## 8.0 ON SITE NEUTRON BASED ASSAY EXPERIMENTS

In order to confirm the predictions of the simulations, the most promising neutron simulation configurations were duplicated using actual fuel. The results of these tests are given in the following sections.

### 8.1 EXPERIMENTAL SETUP

During this phase of the study, an attempt was made to verify the validity of the computer simulations described in section 7.0. Due to constraints in financing and manpower, only the best of the neutron techniques were tested. In reviewing the simulations, one will recall that the Am-Be source provided performance that was comparable to that obtainable with Cf-252. Since it was found that it is possible to characterize the crate with fewer source manipulations by using Am-Be, the Am-Be source was selected as the standard to be used during the experimental portion of this study.

The first step in the experimental procedure was to identify a useable source. This step involved the location of a source that would provide a reasonable neutron count rate when applied to the worst case configuration as defined in section 7.0. Again, a review of the results obtained earlier will indicate that the overall sensitivity of the detector is of the order of .00001 counts/source neutron. The simulations indicated that minimum count rate deviation that can be expected to occur in the event of a deficient fuel crate is  $\pm 10\%$ . If we now assume that we wish to obtain a confidence level of 90% in the collected data (excluding errors in positioning and setup variations), a maximum gross statistical uncertainty of 1% is required. To achieve this

statistical error the total integrated count that must be collected by the detector is 10,000. If we now assume we have a maximum of 10 minutes in which to characterize a fuel crate by taking 6 simultaneous measurements, we find that a minimum source strength of 10 million n/s is needed. In order to provide this output, a source activity of 10 Ci is required. During the experiment a 10 Ci Am-Be source was used.

Once the proper source was located, the next step required for the experimental verification of the simulations was the selection of the proper detector and electronics. The detector selection was based on availability. This resulted in the use of a one inch diameter, ten inch long, four atmosphere He3 counter. This counter is sufficiently small so as to allow the positioning of the detector in the skid area of the fuel crate. The minimum clearance in this area is four inches. For the neutron experiments, the detector was placed on the top of the fuel crate and symmetrical behaviour was assumed to be applicable for the bundles on the bottom of the crate.

The gas filled detector that was used requires a charge sensitive amplifier and a 1300 V power supply. In addition to the power supply, a shaping amplifier, SCA scaler and timer were required to condition the signal and to count the neutrons. The block form of the final configuration is shown in figure 22. All functions required for the operation of the detector were provided by a modified Picker nuclear timer/scaler module, model spectroscaler IIIA. The system was calibrated by setting a lower level discrimination window to pass events that exceeded  $1/2$  the Q energy of the neutron - He3 reaction (746 KeV). By setting the LLD in this way, only neutron induced events were counted.

Once the source and detector were positioned and calibrated, the experiment was started. The results of these experiments are discussed in the

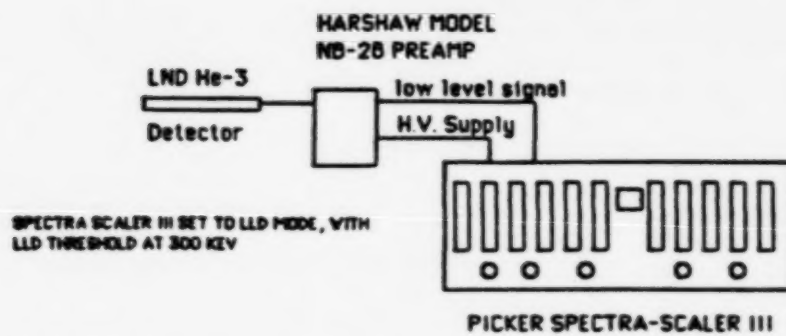


FIGURE 22 NEUTRON EXPERIMENTAL SETUP

following sections.

### 8.1.1 FULL CRATE

In order to obtain a base line count rate for use in missing bundle studies, a series of measurements were taken with the full fuel crate. During this phase, a total of three measurements were taken, one for each position on one end of the crate. The various results are tabulated in table 8.1.1.

Table 8.1.1. Results from full crate Am-Be Neutron measurements

| Source position | Missing bundle position | CPS     | Normalized experimental results | Normalized simulation results |
|-----------------|-------------------------|---------|---------------------------------|-------------------------------|
| B1-B2           | N/A                     | 31+/-2% | 1.00                            | 1.00                          |
| B3-B4           | N/A                     | 42+/-2% | 1.00                            | 1.00                          |
| B5-B6           | N/A                     | 30+/-2% | 1.00                            | 1.00                          |

Bundle positions use standard convention described in figure 4.  
Detector position is directly above source on surface of crate.

The tabulated values indicate that the observed count rates are low but are of sufficient magnitude to obtain reasonable counting times.

### 8.1.2 MISSING TOP PLANE BUNDLE

Once a baseline count was obtained for a full fuel crate, bundles were removed from the top plane of the crate and measurements were taken. In order to obtain satisfactory statistics, a counting time of two minutes was used for each position. The resultant data is tabulated in table 8.3. In comparing the results obtained for this experiment with those which were derived through simulation, we find that the difference in detector count for bundle to missing

bundle cases were lower than those predicted . Still, the deviations observed were sufficient to determine that the bundle was missing.

To understand the reason for these differences, one must first recall the characteristics of the detector that was used in the experiment. Examining the interaction cross-section for He3 detection gas will reveal that the cross section is the familiar  $1/v$  response. As a result, the detection process is heavily biased toward the detection of thermal neutrons.

Since the simulations that were carried out modeled only the fuel bundles and made no considerations for the components of the packing crate and concrete floor, the background of scattered neutrons emerging from these materials are not included in the theoretical detector count rate. Since these materials are of high hydrogen content they will tend to quickly down grade (moderate), the neutron energy. Also, since the scattering bodies retain the same geometry regardless of whether or not a bundle is present, the observed count rate will be elevated in both instances. Since the missing bundle contributes a fixed number of neutrons to the count rate the total observed deviation will be less when the moderating materials are present than it would have been if the moderating materials are ignored. Even though the moderating effect of these materials is small, their contribution to the total observed count rate may be high due to the fact that moderated neutrons are more readily detected by the detector. In order to minimize this effect, one would have to either reduce the amount of moderating material or decrease the low energy detection cross section for the detector. Of these possibilities, the more feasible is the reduction of the low energy cross section of the detector. This can be accomplished by surrounding the detector with selected thermal neutron absorbers. While this possibility is recognized, it was not pursued due to time

constraints. It is suggested that this possibility be investigated should the decision to proceed with the neutron based FFV technique be made.

### 8.1.3 MISSING CENTER PLANE BUNDLE

A test was also carried out to test the validity of the mid plane detectability predictions. The experimental setup and counting times used in these tests were identical to those used in the top plane tests. The results from these tests indicate that the deviations observable for missing bundles was less than that which was predicted. As in the previous case, the observed deviations were sufficient to determine that there was a fuel bundle removed. Table 8.3 documents the results obtained from these measurements.

## 8.2 CONCLUSIONS REACHED THROUGH EXPERIMENTATION

The results obtained from the various experiments are tabulated below. From these we find that several conclusions can be reached in regards to the performance of the neutron based FFV technique. These are:

1. The neutron technique is capable of detecting missing bundles with a minimum certainty of  $9\% \pm 3\%$ .
2. Count rates are sufficient so as to allow collection of usable signals in 2 minutes per detection/source site.
3. Overall neutron efficiency is low (.00001 times)
4. A total of 12 measurements per crate are needed to fully characterize a full fuel crate. During this characterization, two measurements can be made simultaneously thereby reducing the total number of measurement steps

required 6. Statistical error obtainable with nominal counting times of 2 minutes is of the order of 2%.

### 8.3 TABULATED RESULTS

Table 8.3 contains a summary of the results obtained from the neutron based FFV experiments. Also included in this table are the simulation results obtained from the computer modeling of the experimental setup. The values have been obtained with a counting time of two minutes using a He3 detector and Am-Be neutron source. Normalized results refer to the difference between a full fuel crate and a crate known to be deficient in by one bundle. The experimental measurements have been undertaken for one quarter of a fuel crate as it is assumed that symmetry will ensure the other quarters will yield identical results.

Table 8.3. Summary Of Experimental And Theoretical Results Obtained With Am-Be Neutron Source

| Missing Bundle Position | Source Position | Normalized Experimental Result | Normalized Simulation Result | Difference Detectable |
|-------------------------|-----------------|--------------------------------|------------------------------|-----------------------|
| A1                      | B1-B2           | .89 $\pm$ 3%                   | .80 $\pm$ 5%                 | Yes                   |
| A2                      | B1-B2           | .77 $\pm$ 3%                   | .71 $\pm$ 5%                 | Yes                   |
| A3                      | B3-B4           | .82 $\pm$ 3%                   | .70 $\pm$ 6%                 | Yes                   |
| B1                      | B1-B2           | .91 $\pm$ 3%                   | .87 $\pm$ 7%                 | Yes                   |
| B2                      | B1-B2           | .79 $\pm$ 3%                   | .63 $\pm$ 7%                 | Yes                   |
| B3                      | B3-B4           | .71 $\pm$ 3%                   | .56 $\pm$ 7%                 | Yes                   |

## 9.0 APPLICABILITY OF IAEA EQUIPMENT TO FRESH FUEL VERIFICATION TECHNIQUES

In reviewing existing IAEA equipment, we find that equipment requirements for both the neutron and gamma verification techniques can be partially fulfilled with currently available equipment. The various equipment requirements for the FFV techniques discussed in this report have been summarized below.

| TECHNIQUE | EQUIPMENT REQUIRED                                                       |
|-----------|--------------------------------------------------------------------------|
| Neutron   | He3 detection*, H.V. supply, Am-Be source*, amplifier, SCA, and counter. |
| Gamma     | Collimator, Cd-Te detector*, amplifier, L.V. bias supply*, and MCA.      |

In both cases the equipment not available from IAEA inventories is denoted with a \*. Where the equipment required for the technique is available from standard inventory, the most likely candidate for use has been selected and summarized below.

| TECHNIQUE | EQUIPMENT REQUIRED                          | MOST LIKELY IAEA DEVICE TYPE |
|-----------|---------------------------------------------|------------------------------|
| Neutron   | H.V. supply, amplifier<br>SCA, and counter. | Silena BLS-27                |
| Gamma     | Amplifier, and MCA.                         | Davidson MCA/amplifier       |

In the case of non standard equipment, the reader is referred to section 10 of this report.

## 10.0 IMPLEMENTATION OF VARIOUS TECHNIQUES

Due to the specific nature of the proposed FFV techniques as well as the time and manpower constraints placed on IAEA personnel at the time of the FFV execution, the procedures to be followed in order to carry out the proposed techniques must be made as simple as possible.

This section of the report details the equipment and methodology that is felt as being best suited to in field application of proposed FFV techniques.

### 10.1 IN FIELD EXECUTION OF THE GAMMA BASED FRESH FUEL VERIFICATION TECHNIQUE

The gamma technique has to its advantage the fact that the technique is completely passive. As a result, it is only required that the operator measure gamma flux at the appropriate points and record the data for future assessment. If post processing of data is to be used, it is highly critical that the gamma flux measurements be carried out with a high degree of accuracy as there is little opportunity to recollect the data after the inspector has left the facility. As a result, while developing a methodology for in field application of the gamma based technique, the requirement of accurate measurement has been of paramount concern. Of secondary concern has been the minimization of data acquisition time which results in the more productive use of manpower and a reduction of gamma dose received by IAEA personnel.

#### 10.1.1 COLLIMATION

The first step in developing an easy to implement FFV technique, is

the development of a suitable collimation system. Since it has earlier been determined that only internal crate scanning is of use for FFV, it becomes immediately obvious that a small diameter collimation and detection system must be used. Since two types of internal scanning have been investigated, each with special collimation requirements, two schemes have been devised. These are discussed in the following sections.

From the data collected from the simulations and experiments it has been found that the flux remains essentially constant along the length of the bundle. As a result the missing bundle determination can be made at any point along the bundle length with optimal results being obtained from midpoint measurements. This opens the possibility that a single point measurement can be taken for each bundle, and in fact most tabulated data presented so far is based on this method. Since only a single position must be monitored it is possible to access the fuel crate from the side exposed to the access corridor. In this orientation, only the side view of the bundles are visible but with the proper channels provided a probe type collimator could be positioned to lay midway between any two bundles. This arrangement would permit the verification to be carried out without having to move the fuel crate to gain access to the side. Even with such a scheme it should be noted that the first crate would still have to be moved in order to gain access to the rear crate.

#### 10.1.1.1 FOUR BUNDLE COLLIMATION SCHEME (UNGATED COLLIMATOR)

The first internal measurement scheme to be studied is based on the ungated collimator described in section 5 and 6. As one will recall, the counting time required to verify a group of four bundles is in order of one to

two minutes. This measurement would have to be repeated a total of 12 times if a complete crate is to be verified. In order to reduce the full crate verification time, a simultaneous acquisition technique could be used. To realize such a technique, a multiple detector array would be needed. This multiple detector arrangement would consist of two ungated collimator/detector assemblies positioned within a rigid tube. Each detector assembly would consist of a 360 degree collimator and detector.

The use of this multi detector fixture would entail the insertion of the assembly into an opening between the fuel bundles in such a way as to position the detectors in the center of the two bundles. This positioning could be such that the probe is parallel or perpendicular to the bundle depending on which side is visible. Assuming all detectors are matched in terms of efficiency, the operator could make a rapid assessment as to the presence of a bundle by comparing a reference to the count obtained from each detector. Any deviations in the expected count would alert the operator to a possible missing fuel bundle. A drawing of the proposed two detector assembly is shown in figure 23.

In operating the proposed FFV equipment, in a non-enrichment discrimination mode, we find that gamma flux at the internal detector site is of the order of 10,000 gamma rays/cm sq./second. Assuming that we are using a small Cd-Te detector with an efficiency normalized to surface area of 3%, we find that a count rate of the order of 300 CPS can be expected. Since we must detect a missing bundle that is expected to result in a deviation in flux of 9%  $\pm$  5%, we find that given the statistical limits of measurement, a counting time of approximately 20 seconds would be required to characterize eight bundles with an error of  $\pm$  1%.

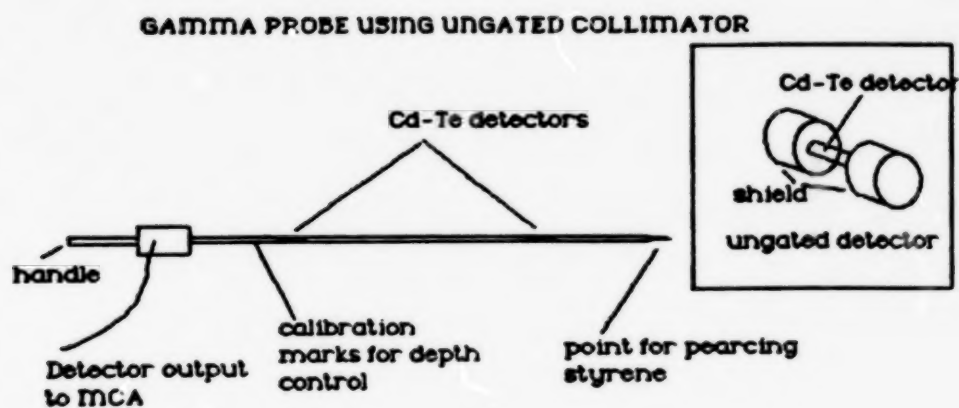


**FIGURE 23 - POSSIBLE MULTI COUNTER SYSTEM  
UTILIZING 360 DEGREE INTERNAL  
COLLIMATOR.**

No account of base or full crate reference data statistical errors are assumed as this data would be originally collected using long counting time (and therefore should show low statistical error). The actual position of the probe would have varying effects on the accuracy of the technique, depending on the orientation of the probe axis. In the case where the probe is parallel to the bundle, probe misalignments of up to 30% of the probe length would have minimal effect on the observed count rates due to the flat flux profile along the bundle length. In the case where the probe is perpendicular to the bundle, probe position will be extremely critical with the requirement that the probe be within about .3 cm of its proper location. This could be achieved by marking the probe with accurate calibration marks.

Examination of figure 23 will indicate several features of the design. The first is that the entire probe would consist of a 1 to 2 cm diameter rigid low cross-section tube (thin walled 7075 aluminum) that could be pushed through the fuel crate. The small size of the Cd-Te detectors would enable such a small diameter probe to be constructed. The probe would also be marked with calibration lines spaced in bundle intervals so as to ensure proper positioning within the crate. Detector bias and signal would be transmitted to the power supply and amplifier via a small diameter (RG-174) coaxial cable. The use of this small cable is made possible by the low bias voltage required by the detector. Finally, the two channel amplifier (one amplifier per channel) would be affixed to the end of the probe and would also function as a handle. Power to the entire assembly would be provided by the Davidson MCA. Spectrum collection from each individual detector would be selectable by a switch which could select one, or two detectors.

While it has been shown that it is possible to readily detect a

deficiency of most bundles through the use of the ungated detector it should be noted that this collimation scheme has been found to produce only marginal deviations for some bundle locations. In recognition of this, it is recommended that the ungated detector not be adopted as a FFV device unless cost is an overriding consideration. Instead it is recommended that if a gamma based technique is to be adopted, the gated collimator system described in the following section be used.

#### 10.1.1.2 SINGLE BUNDLE COLLIMATION SCHEME (GATED COLLIMATOR)

A second type of internal measurement has been studied where an internal detector is used to characterize a single fuel bundle. This scheme, while taking more time to analyze a fuel crate, would allow the verification of a single bundle. If we again assume a non-enrichment type of measurement and use the Cd-Te detector described in 10.1.1, we find that a counting time of the order of 60 seconds is required to obtain an accurate flux value for each bundle. Since this measurement must be repeated for each bundle, the total time required to characterize a full crate is prohibitive unless some form of parallel measurement is used. In recognition of this fact, a scheme similar to that detailed in section 10.1.1.1 is proposed. Such a scheme is illustrated in figure 24. This device shares most of the attributes of the previous detector with the exception that it uses a total of eight detectors.

Radial positioning of the gated detector becomes a major concern since misalignment of the collimator can severely corrupt the measured flux values. To minimize the probability of radial misalignment occurring, the operator would be required to mark the vertical centerline of the fuel bundle mid plane and

Cd-Te detector epoxied in place

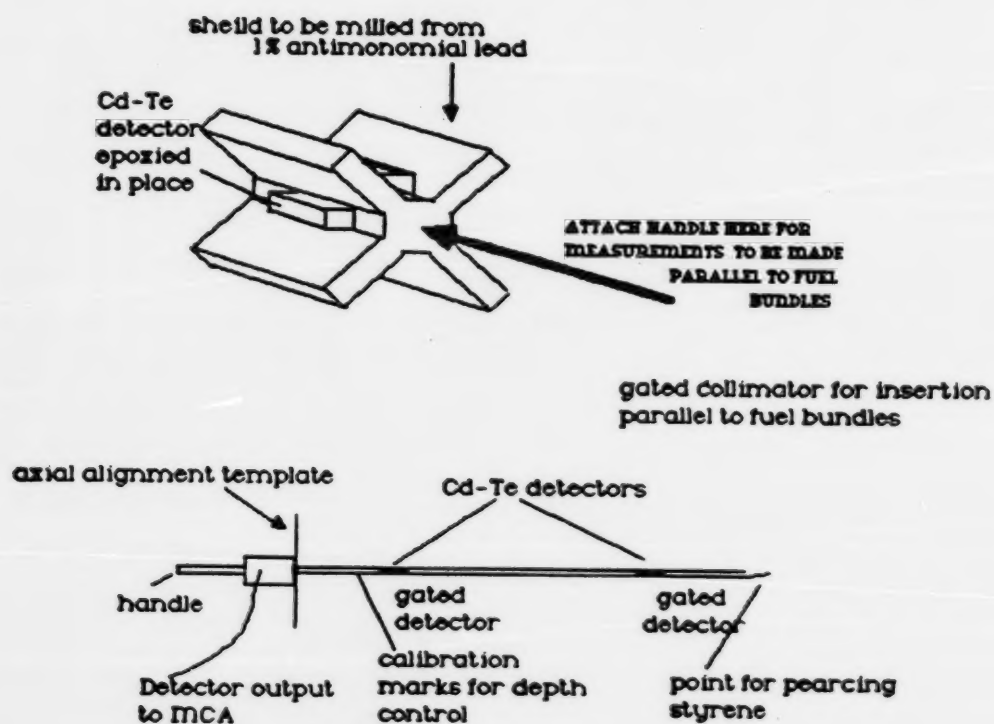


**FIGURE 24 - POSSIBLE MULTI COUNTER SYSTEM  
UTILIZING GATED INTERNAL  
COLLIMATOR SHOWING PARALLEL  
COLLIMATOR DESIGN**

position the probe so that its detection mid plane corresponded. Since there is little room for error in this alignment procedure, the probe would be a rigid tube with an alignment guide affixed. The diameter and length of the tube would be similar to that proposed in section 10.1.1.1. The comments of section 10.1.1.1 regarding tube insertion length misalignments also apply in this case, thus in the worst case an estimated tolerance of  $\pm 0.3$  cm could be tolerated. This figure would be achieved by also using the radial alignment template as a fixed depth gauge.

Different probes would be needed for the perpendicular and parallel insertion direction. These differences would be in the collimation system only, example of which are shown in figure 24. The fabrication of the probes would entail the purchase of bare small diameter Cd-Te crystals and mounting the crystal as an integral part of the collimator.

By using the estimate derived in 10.1.1.1 and accounting for the one quarter count rate because of the collimator design (ie. looking at only one bundle instead of four), we find that a counting time of 40 seconds is required in order to obtain a statistical uncertainty of 2%. By counting eight bundles in parallel we find that a complete fuel crate characterization is feasible within 10 minutes.

#### 10.1.2 FUEL CRATE CHANGES

In both the case of gated and non-gated internal flux measurement, a channel between each group of 4 fuel bundles is required. Fortunately, given the existing structure of the fresh fuel shipping crate, is is a simple exercise to incorporate such a passage into the packaging material.

By introducing a segmented channel in the slab section of the spacer assembly, it should be possible to provide a passage through the crate while retaining most of the slab strength. A section of the proposed channel construction is shown in figure 25. During insertion, the probe would be required to break through the channel ends. To do this, the probe would have to be fitted with a sharp point. There would be minimal chance of damaging the bundles with this point as each side of the styrene foam is surrounded by a sheet of corrugated cardboard.

The use of a segmented channel in the slab sections of the packing crate are unavoidable if one wishes to avoid problems when handling the sheets during fuel packaging. It is felt that handling problems will occur since the introduction of a continuous slot in the polystyrene slab would create a very weak point in the structure. By adopting a gated channel, some of the original strength would be retained while providing a reasonably low resistance passage for a probe to pass through the crate. To gain access to the channel, the person executing the FFV procedure would be required to cut out an access hole in the cardboard wall of the crate. This could be accommodated through the use of a template, preferably marked on the crate.

The direction of the channels could be both parallel and perpendicular to the bundles so as to allow probe insertion from either direction. It should be noted that each orientation would require a different probe. The channel spacing would be such that the inspector could select any of a number channels in the perpendicular orientation so as to minimize the chance of a partially missing bundle (half or quarter length) being missed.

There exist differences in the crate designs used by different fuel manufacturers. It would be required that all manufacturers adopt a common

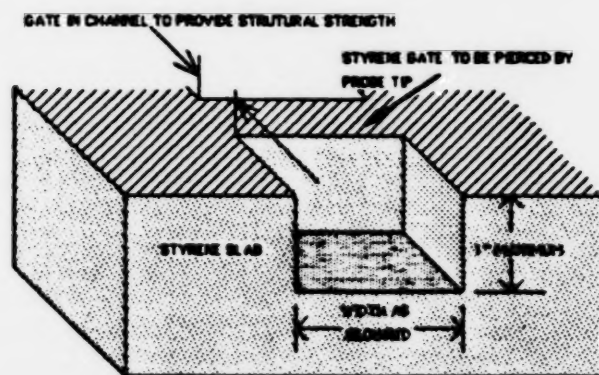
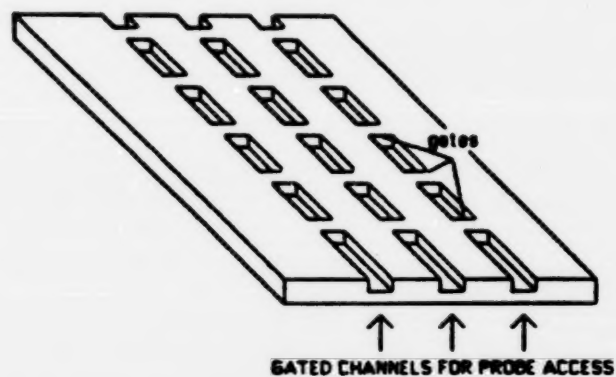


FIGURE 25. PICTORIAL OF PROPOSED GATED CHANNEL DESIGN

design in which the bundle to bundle spacing is maintained constant across the fuel crate. Currently General Electric packing crates do not possess this property.

#### 10.1.3 DOSIMETRY CONCERNS - GAMMA TECHNIQUE

During the course of the FFV measurement, the IAEA personnel will receive a certain radiation dose. Although radiation exposure is expected during on site visits, it should be minimized where possible. This section derives the expected dose that the typical operator will receive for each crate of fuel that is verified.

First we take the average of the various measured background radiations. From the data collected on site we find that this average value is 8 mr/hr (section 3.1.2). Next, taking an average eight bundle measurement time of 60 seconds and multiplying by the number of samples required to characterize a full fuel crate ( 6 samples) (sections 5.3,10), we find that the total procedure will require that the operator receive a dose of 1.3 mr per crate. Further, assuming that the crate is to be removed from the shelf and that this procedure will require an additional 300 seconds, this will add an additional dose of .7 mr, resulting in a total gamma dose of 2 mr/crate.

Note that there is no neutron dose associated with gamma based FFV technique as the only neutron exposure would come from background, which in the case of the areas surveyed is 0 mR/hr.

## 10.2 NEUTRON BASED FRESH FUEL VERIFICATION

As discussed in earlier sections, the neutron based technique requires that the source be placed within the fuel crate in a fashion similar to that used in the gamma based techniques. Once this positioning is completed, the FFV procedure can be carried out from the crate exterior. This section details how this measurement could be carried out and describes the most likely basic design of the equipment.

### 10.2.1 SOURCE GEOMETRY

As indicated in the prologue to this section, the source used in the neutron based measurement must be placed within the fuel crate. This can be accomplished by attaching the neutron source to a probe similar to the one used in the gamma technique. In this case, the source would be encased in a threaded section that would be attached to the rod. The operator would then push the rod into the crate to the proper position as indicated by calibration marks on the probe body.

Once the measurement is completed, the source can be unscrewed from the probe and stored in a transportation flask for the transit to the next assay site.

The specifications of the source have been reviewed earlier but are recalled here for completeness. The various source parameters are given in table 10.2.1.

Table 10.2.1- Neutron Source Specifications

|                         |                   |
|-------------------------|-------------------|
| Source type             | Am-Be             |
| Source activity         | 10 Ci             |
| Diameter                | Maximum of 1.0 cm |
| Neutron dose rate at 1m | 22 mR/hr.         |
| Gamma dose rate at 1m   | 25 mR/hr.         |
| Neutron output          | 22,000,000 n/sec. |

The selection of the source strength was made by examining the experimental data presented in section 8 and selecting a source strength that would give a counting time of 120 seconds for a statistical uncertainty of 3%.

#### 10.2.2 EQUIPMENT REQUIREMENTS FOR NEUTRON TECHNIQUE

In addition of the source, a detection system is required. In the actual application of the FFV neutron technique, the equipment configuration would likely approximate that shown in figure 26. Here, the source probe would be as described in the previous section, the detector would consist of a large area He3 detector and a detector bias supply. Amplifier and scaler would consist of IAEA equipment as discussed in section 9.0.

The procedure to be followed during a verification would involve the positioning of the source and detector to the measurement position, and initiating data collection. Positioning of the detector would be aided by the use of a template either printed or placed on the top of a fuel crate. If a template is utilized it would likely consist of a thin plastic sheet marked so as to indicate the proper locations for the detector. Once the measurement was complete, the source would be withdrawn and returned to the appropriate storage container.

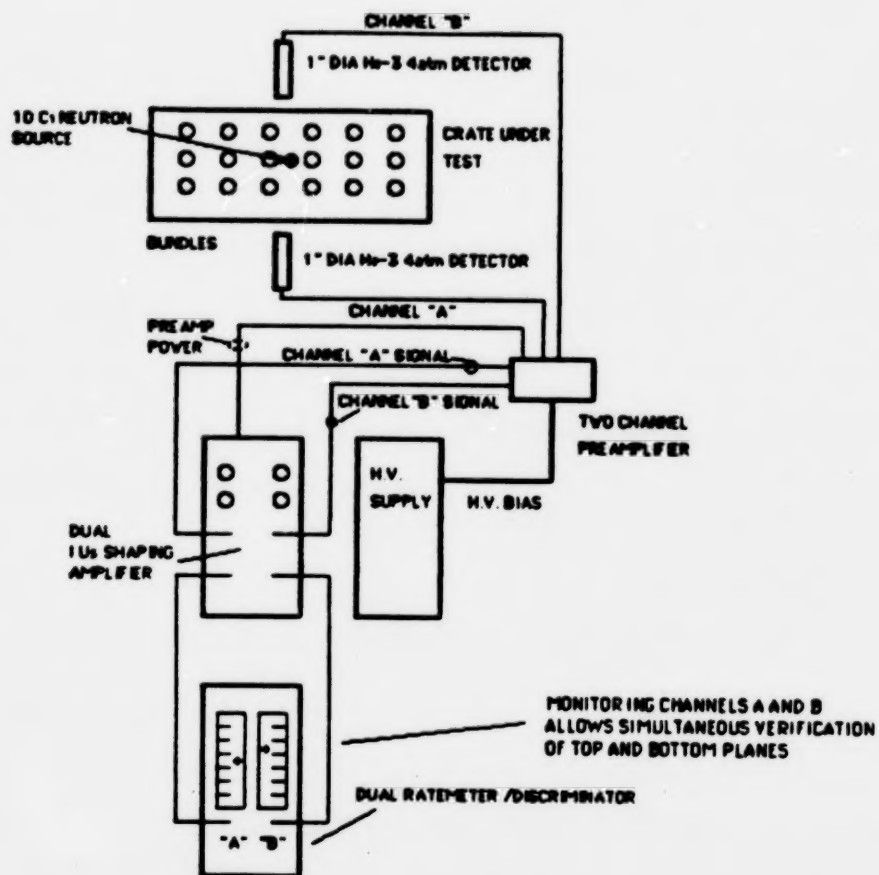


FIGURE 26. PROPOSED NEUTRON BASED MEASUREMENT CONFIGURATION

### 10.2.3 FUEL CRATE CHANGES

Since the neutron source probe must be located in the mid plane of the fuel crate for proper operation an access channel would have to be provided. In this case the access channel would have to be built into the bundle spacer assembly to allow positioning of the source in the mid plane position. Aside from this difference this channel would be constructed in an identical fashion to that proposed for the gamma technique (section 10.1.2). It is felt that differences in the fuel crate design would not greatly affect the performance of the neutron technique, as such it is felt that no redesign would be required in the case of the General Electric crates.

### 10.2.4 DOSIMETRY CONCERNS - NEUTRON TECHNIQUE

Unlike the gamma based FFV technique, the neutron technique utilizes neutrons during the measurement procedure. As a result, it is important to recognize and characterize the quantitative aspects of the additional hazard presented by the presence of the neutron source.

To calculate what exposure the operator of the FFV device would receive, we simply integrate the dose rate produced by the source over the measurement interval. In carrying out this calculation, we first refer to section 10.2.1 and obtain the dose rate produced by the source. Next, we obtain an estimate of the time the operator will be exposed to the source by assuming a 5 minute set-up time for the positioning of the source and the detector. Once the source is positioned, we then reduce the exposure from the source by the known escape probability for the source gamma and neutrons. This last value can be obtained

from the simulations section of the report. Once this effective dose rate is obtained, we then integrate over the total measurement time of 720 seconds. The results of these calculations are tabulated in Table 10.2.4.

Table 10.2.4 - Neutron based FFV radiation exposure estimate

|                                                           |                                              |
|-----------------------------------------------------------|----------------------------------------------|
| Exposure from bare source                                 | gamma - 25 mR/hr.<br>neutron - 22 mR/hr.     |
| Time exposed to bare source<br>per crate of fuel verified | 300 seconds                                  |
| Time exposed to source in crate                           | 720 seconds                                  |
| Dose rate from source in crate                            | gamma - .002 mR/hr.<br>neutron - .002 mR/hr. |
| Total integrated dose from source                         | gamma - 2.1 mR<br>neutron - 1.85 mR          |

In addition to the dose from the source we must add the normal dose obtained from background gamma. This value can be obtained from section 10.1.3 and when added to the source dose, results in the following.

|                                 |           |
|---------------------------------|-----------|
| Gamma dose per crate verified   | - 4.1 mR  |
| Neutron dose per crate verified | - 1.85 mR |

It should be noted that this is a worst case value as it is assumed that the operator would be in very close proximity to the crate during the measurement. In actual fact he could easily reduce his dose by standing away from the point of measurement.

## 11.0 COMPARATIVE ANALYSIS OF RADIATION BASED FFV TECHNIQUES

The various study results have been tabulated below.

|                                                | Neutron      | Gated collimation<br>gamma | 360 degree collimation<br>gamma |
|------------------------------------------------|--------------|----------------------------|---------------------------------|
| Crate penetration<br>required                  | yes          | yes                        | yes                             |
| Partial bundle<br>deficiency detection         | yes          | no                         | no                              |
| Substitute material<br>detection               | yes          | yes                        | yes                             |
| Bundles per set                                | 6 effective  | 8 effective                | 8 effective                     |
| Estimated counting<br>time per bundle set      | 120 sec      | 40 sec                     | 20 sec                          |
| Set-up time                                    | 300 sec      | 300 sec                    | 300 sec                         |
| Missing bundle<br>detection threshold          | <1<br>bundle | 1<br>bundle                | 1<br>bundle                     |
| Statistical<br>Counting error                  | 2%           | 2%                         | 2%                              |
| Possible application<br>to enrichment analysis | no           | yes                        | yes                             |
| Operator neutron dose<br>per crate verified    | 2 mR         | 0 mR                       | 0 mR                            |
| Operator gamma dose<br>per crate verified      | 4.1 mR       | 2 mR                       | 2 mR                            |
| Radioactive source<br>required                 | yes          | no                         | no                              |
| Measurement possible<br>while crate on shelf   | no           | yes                        | yes                             |
| Transportability                               | Radioactive  | easy                       | easy                            |
| Non-standard equipment                         | none         | probe                      | probe                           |

## Legend -

|                        |                                                                                                                                                                                                                                         |
|------------------------|-----------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------------|
| Set -                  | Number of bundles tested per measurement using techniques described in chapter 10. It should be noted that six sets of measurements would be required for both the gamma and neutron techniques to verify a full crate of fuel bundles. |
| Statistical Error      | Counting Statistical error likely to be encountered using specified counting time.                                                                                                                                                      |
| Non-standard Equipment | Type of equipment not commercially available but required for technique.                                                                                                                                                                |

A review of these results will indicate the relative merits of each technique. From these results, it is obvious that all techniques are feasible. As a result, it is left to the IAEA to determine which technique is most suitable for use as a fresh fuel verification tool.

## 12.0 RECOMMENDED COURSE OF ACTION

This report details the operating procedures as well as the expected performance that can be obtained from several types of fresh fuel verification schemes. The next step in development is the determination of an operating methodology and subsequent development of suitable equipment that can be used by the IAEA.

Finally, it should be noted that while the section relating to technique development outlined several possible final configurations, it may be found that these configurations are for some reason unacceptable. As such, the data presented in section 10 should only be considered as a guide to the performance that can be expected from the various techniques in a specific operating environment. Appropriate adjustments should be made in the expected performance if significant changes in the operation or design are undertaken.

## 12.1 PROJECTED COSTS TO IMPLEMENT EACH TECHNIQUE

Table 12.1.1 indicates anticipated costs required to implement the various gamma based fresh fuel verification techniques.

Table 12.1.1 - Anticipated Costs of Gamma Technique

| Item                     | Cost        | Notes                                                                                                |
|--------------------------|-------------|------------------------------------------------------------------------------------------------------|
| 1) Multichannel Analyzer | \$10,000.00 | - Only one required for many detectors                                                               |
| 2) Probe & Collimator    | \$ 3,000.00 | - Does not include price of detectors<br>- Indicates cost of precision machinery for two collimators |
| 3) Detector Power Supply | \$ 1,500.00 | - Only one required for many detectors                                                               |
| 4) Cd-Te Detectors       | \$ 1,700.00 | - Cost per detector. Multiply this figure by number of detectors                                     |
| 5) Preamplifier          | \$ 500.00   | - Cost per detector. Multiply this figure by number of detectors                                     |
| 6) Shaping Amplifier     | \$ 600.00   | - Cost per detector. Multiply this figure by number of detectors                                     |

Note that the costs of items 3, 4, 5 and 6 are per detector, and that it is possible to apply the gamma technique with only one detector channel. This of course requires that the measurement time be extended if a full fuel crate is to be verified. These prices reflect current commercial price lists and should only be taken as estimates.

Table 12.1.2 - Anticipated Costs of Neutron Technique

| Item                         | Cost       | Notes                     |
|------------------------------|------------|---------------------------|
| 1) High Voltage Power Supply | \$1,600.00 | - One required per system |
| 2) He-3 Neutron Detector     | \$1,500.00 | - Two required per system |
| 3) Preamplifier              | \$ 700.00  | - Two required per system |
| 4) Dual Amplifier            | \$1,300.00 | - One required per system |
| 5) Ratemeter Discriminator   | \$ 800.00  | - Two required per system |
| 6) 10Ci Am-Be Neutron Source | \$ 8940.00 | - One required per system |

Note that the costs of items 2, 3, 4 and 5 are per detector. In a minimum configuration, one detector can be used with a corresponding increase in fuel crate verification time.

Table 12.1.3 - Fuel Crate Changes

| Item                                     | Type             | Cost        | Notes                  |
|------------------------------------------|------------------|-------------|------------------------|
| 1) Change crate for equal bundle spacing | General Electric | \$30,000.00 | - Retool insert mould  |
| 2) Insert gates in spacer sheets         | All              | \$ 8,000.00 | - Modify existing tool |

Note that these costs are based on minimum information estimates obtained from the Tool and Die Industry and can only be taken as a rough guide.

## A-1 GAMMA PHYSICS TUTORIAL

As one will recall from the brief review given earlier, both the geometry and enrichment of fresh fuel bundles can be determined through spatial mapping and energy selective gamma monitoring of the fuel crate. The purpose of this section is to familiarize the reader with the physics that predict gamma behaviour as well as how these concepts can be applied to the task of fresh fuel verification.

The first concept that will be covered is that of gamma imaging. As a first step we must consider fuel bundles that are to be tested as discrete cylinders of gamma emitting material. If we confine our view of one these cylinders to a small area on its surface, and measure the gamma flux emitted through that surface, we can obtain a measure of point activity. If we make a series of these measurements we can then construct a map of gamma flux, which, when presented in graphical terms, should closely match the outline of the gamma emitting material (which in this case is the fuel bundle). This process is illustrated in Figure 1.

While these series of measurements may under certain conditions provide an indication of the position of the fuel bundles within the fuel crate, the confident application of this concept requires that we have a more in depth understanding of how these measurements are affected by different radiological and geometric conditions. If we consider the hypothetical flux measurement carried out in the previous paragraph, we find that the gamma flux measured at the imaginary surface is actually related to the activity contained in a volume bounded on one side by the area that one is viewing. In the most simple case, this volume is bounded by the measurement surface on one side and extends to an

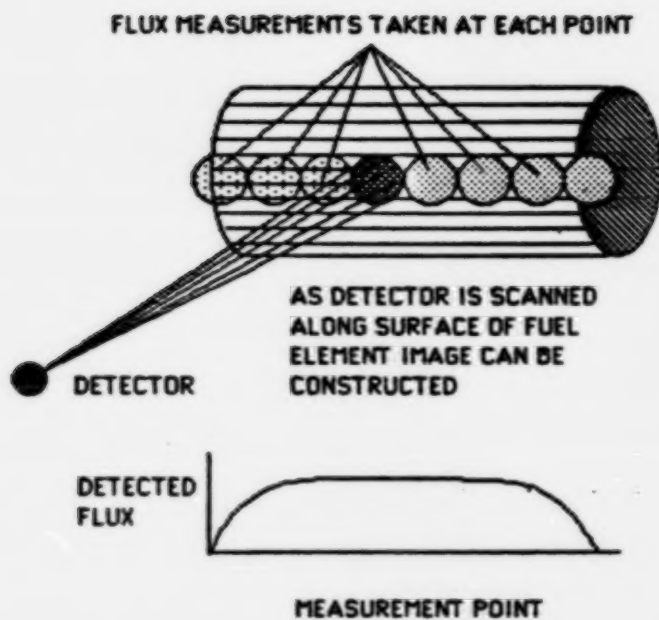


FIGURE 1. GAMMA IMAGING USING COLLIMATED DETECTOR

infinite depth. Although these bounds are theoretically infinite in depth, practical considerations limit this depth to a relatively small value. The magnitude of this effective length is governed by several factors, the most important of which are the far field behaviour of gamma photons and the physical properties of emitting material (which in this case is a fuel bundle). This process is illustrated in Figure 1.

Of the factors that affect the effective depth of the measurement volume, the most simple to understand and characterize is that which relates to the far field behavior of gamma flux. Simply stated, the measured flux from an isotropic point source in the far field environment is given as:

eq. 1

$$I(r) = \frac{I(0)}{4\pi r^2}$$

where:  $I(r)$  = detectable flux at a point  
which is distance  $(r)$  from  
point emitter

$I(0)$  = total flux emanating from  
the point emitter

If we now place an absorbing medium between the detector and source, after simple mathematical operations we arrive at the well known attenuation equation.

eq. 2

$$I(x) = I(0) e^{-Zx}$$

where  $I(x)$  = uncollided gamma flux  
reaching a point which  
is a linear distance  $x$   
through an attenuating  
medium.

$I(0)$  = initial gamma flux  
emanating from a source  
measured at the point  
of the source.

$Z$  = absorption cross section  
of the attenuating media.  
This property is a function  
of molecular and electron  
density of a material.

If we apply the cross-section values for UO<sub>2</sub> to this equation we find that magnitude of this equation is far less than that of equation number 1. As a result, we find that the attenuation presented by the UO<sub>2</sub> is the dominating effect when defining the effective volume thickness. Because of this fact, only

the attenuation term will be considered in the discussions that follow.

If we apply the attenuation effect to a practical situation we find that fuel pencils must be considered as gamma opaque objects. Because of this property, the entire concept of gamma imaging of fresh fuel bundles is greatly impaired since it becomes impossible to see either bundles behind bundles or even pencils in the interior of a bundle. Despite this short coming, the concept of imaging may have some application in the determination of the bulk presence of fresh fuel bundles. Consequently, further study of the physics is warranted.

Continuing the investigation into gamma imaging brings us to the practical application of a gamma based FFV technique. If we extrapolate from knowledge of other practical gamma detection systems we find that while there are numerous factors that could adversely affect the performance of a gamma imaging method for fresh fuel verification the single most important consideration that has not yet been mentioned is that of background contamination of the measurement. This factor is especially applicable to the problem at hand as the FFV process requires that fuel imaging take place in a high background area. The quantification of this problem is discussed in section 6.

In conclusion, three potential problems have been identified in the implementation of gamma imaging based FFV technique. To summarize, we have, inter bundle shielding, inter pencil shielding, and back ground contamination to contend with. In recognition of these problem areas, the investigation process has centered on characterizing the extent and nature of these problems and development of means to reduce their impact. As a peripheral activity, potential problems with a less severe impact on the performance of the technique

were analyzed. This was facilitated through the models developed for the analysis of the primary problems.

In addition to the spatial mapping mode of gamma based assay, there exists an enhanced mode of operation which shows promise as a means of determining the enrichment status of fresh fuel. This enhanced technique is based on selective energy counting, where gamma emissions from the U-235 component are separated from those originating from U-238. In the proposed technique this same measurement would take place without removal of the fuel from the fuel crate.

Since the considerations relating to fuel enrichment are closely linked to those relating to the spatial measurements, it is in theory, possible to carry out both measurements simultaneously. As a result, the study of the enrichment determination technique has from this point on been considered as an integral part of the spatial mapping technique.

## A-2 NEUTRON PHYSICS TUTORIAL

The principles that are used in the neutron based assay technique are quite diverse. In essence, every possible type of neutron interaction is taking place during the FFV measurement. These include absorption, scattering, fast fission, and thermal fission. Through proper modelling, it is possible to predict how these various interactions will combine to yield a net result. In this section of this report, both computer simulations and in field tests have been carried out to determine the applicability of neutrons to the task of fresh fuel verification.

Recalling from earlier discussions, we have found that the operation of the neutron based assay technique relies heavily on scattering and absorption. Due to this, the first step in understanding how the technique operates is to examine the various interaction cross-sections of materials that dominate the test setup.

If we start by examining the neutronic properties of U-238, it is found that the various interaction cross sections vary rapidly as a function of energy. These wide variations can be seen in figure 1. This figure shows the scattering, fission, and absorption cross sections for U-238. If we now consider the setup used in the neutron based assay method, we find that a neutron source is used to flood the U-238 with neutrons. As a first approximation, let us assume that the neutrons emerging from the source are non-energetic with an energy of 1 MeV. If we adopt the classical attenuation configuration of a collimated source, absorption media (U-238) and detector figure 2, we find that the neutrons detected by the detector will simply be the difference between those lost through absorption and scattering in the U-238 and

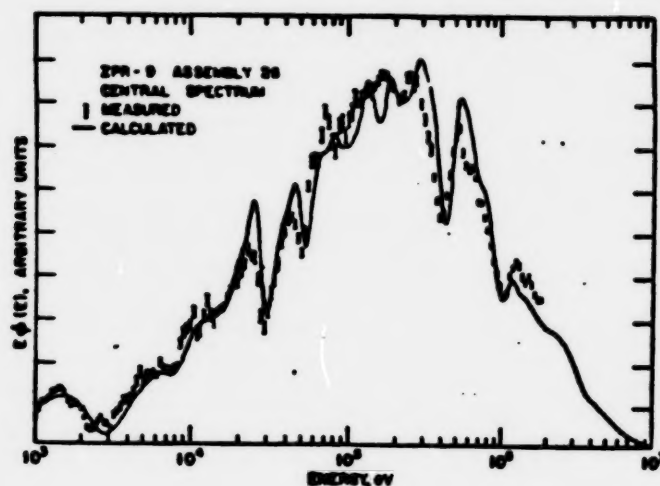
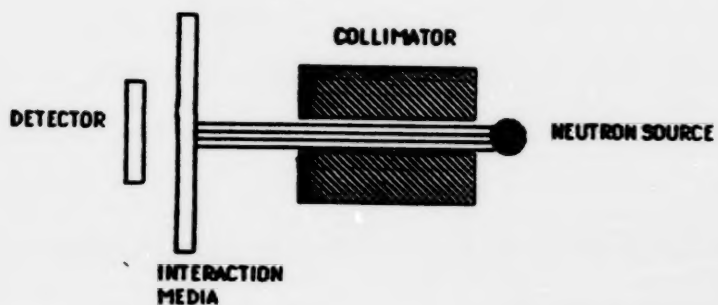


Fig. 1 Central Neutron Spectrum in Mockup of Fast Flux Test Facility Core, Together with Smoothed Fundamental-mode Calculation. ANL Neg. No. 900-308 Rev. 1.

## APPENDIX A-2 FIGURE 2 CLASSICAL NEUTRON ATTENUATION SETUP



the original source strength. To predict the number of neutrons that will be lost in the uranium, we would refer to the cross section plots described previously and calculate the total removal cross section of attenuation mass. The result of this calculation would be the derivation of a numerical coefficient that, when applied to the incident neutron flux value, would yield a value that should be equivalent to the detected flux. While this experimental setup is illustrative, it is very difficult to realize in an actual setup.

If we now adopt an experimental configuration that is more attainable by removing the neutron source collimation and allow the neutron energies to vary over a bounded range, the determination of the neutron flux at the detector becomes more difficult. In order to predict the magnitude of neutron flux reaching the detector, the basic behavioral properties of neutrons must be reviewed. Aside from the basic neutron-nuclear interactions that have already been described in section 4.2, there remains the matter of neutron flux behaviour as a function of distance. In this regard, we resort to the familiar  $1/(r^2)$  equation as a starting point. This equation, which has already been described as it relates to gamma flux, is also applicable to neutron flux. As in the previous case, this equation is only directly applicable in a space that is free of material that interferes with the flow of particles. In cases where there are interfering materials, this equation is modified to an extent dependent on the amount of influence the materials have on the neutrons. In extreme cases where the neutrons have a high probability of interacting with a material, the typical model used to describe the neutron behavior is the single group neutron diffusion equation.

In this equation, the various cross sections are derived through the averaging of the energy dependent cross sections over energies of interest.

While being illustrative of the principles of neutron conservation and diffusion, the single group diffusion equation can at best be considered as a teaching aid which is applicable to very simple situations. To understand why this is so, one must analyze how the neutrons interact with the interfering media. In the case of a fresh fuel bundle, this material is U-238. Recalling from the original discussion on the basics of neutron interactions, we find that we have several types of interactions that can occur. For any given isotope, the probability that these interactions will occur is directly related to the energy of the neutron. One will also recall that one of the primary and most probable interactions is the scattering interaction. Since a result of scattering is a reduction in neutron energy, there will be a definite change in the probability that another neutron-nucleus interaction will occur. This process will continue until either the neutron escapes or is absorbed. As a result of this continuum of reductions in neutron energy, the simple models based on averaged interaction cross sections usually produce erroneous results.

In an effort to adapt analytical tools to the solution of multi energy neutron interaction problems, several schemes have evolved. Of these the most notable is the multi group diffusion model. The premise used in this model is that the various interaction cross sections can be divided into several energy groups, with the cross section of each energy group being derived from the averaging of discrete energy dependent cross sections. While this technique is quite useful in some cases (the Candu reactor core was developed with a two group model), it is totally unacceptable in others. In the case of fresh fuel verification, it turns out that the diffusion model is not applicable to the problem of fresh fuel verifications because of the well known inability of the diffusion equation to handle situations with boundaries possessing greatly

different neutronic properties.

In order to cope with situations such as the one that exists with FFV, tools have been developed which do not rely on analytical assumptions that can be reduced to simple equations. These tools appear in the form of particle tracking, (Monte Carlo), computer codes. These programs are based on tracking of the particle in question, (in this case the neutron), from its origin (emission) to its death (escape or capture). Fairly complex in implementation, the basic concept used in these programs is fairly simple to understand. First, the region under investigation is broken up into cells. Each cell has a specific set of neutronic properties that correspond to material contained within. The source neutron is then released with a defined energy and direction and is allowed to travel until it interacts with one of the cells. After interaction, the new neutron velocity is computed and tracking continues. This process is continued until the neutron is either captured during one of the interactions or escapes from the region of interest. In order to determine what physical interaction occurs at an interaction point, a probability of interaction is computed. This computation is based on a weighted random number (thus the name Monte-Carlo) selection so as to preserve the random nature of the event. By tracking and computing interaction probabilities in this way, the computer mimics natural processes and thereby retains a high degree of applicability. Because of this applicability to a large number of situations, this type of simulation is ideal for use in problems such as those encountered in the current FFV case.

If we now return to the basic physics involved in neutron interaction with the fresh fuel crate, we find that we have a large number of poly energetic neutrons that undergo capture and scattering. Since the scattering results in a

continuum of energies there will be varying numbers of neutrons absorbed at these energies. As a result, there will be a specific "finger print" of absorption and scattering that corresponds to the various cross sections of the fuel. If this neutron spectrum "finger print" is then analyzed it should be possible to determine the quantity and isotope contained in a test sample.

In the case of FFV, the test sample consists of a negligible cross section fuel crate containing massive natural UO<sub>2</sub> fuel bundles. If we irradiate the crate with a neutron source and detect the emerging spectrum, we should theoretically sense the characteristics of the UO<sub>2</sub> fuel. In addition to the normal absorption and scattering processes that occur in uranium, we also find that a significant amount of fission is possible. With natural uranium we find that two forms of fission are possible. The first form is the well known thermal fission of U-235. In this case, we require thermal energy neutrons to initiate the reaction. The second type of fission to occur is the less common fast fission which occurs in U-238. Also, unlike thermal fission, fast fission requires that the initiating neutron possess energies in excess of 1 MeV.

Both thermal and fast fission processes will contribute to the number of neutrons emerging from the fuel crate. Similarly, the neutron energy spectrums emerging from the fuel crate will be modified by these two fission neutron sources. Of particular importance is the fact that the degree to which the spectrum of the neutrons emerging from the crate are modified, will be directly dependent on the magnitude of the fission source term of the material under test. This point is directly applicable to the determination of the enrichment status of the fuel. An increased U-235 enrichment will increase both the thermal fission yield and the number of high energy neutrons emerging from the crate that is under going thermal neutron irradiation.

By combining the neutron spectrum fingerprint and fission source effects, it is hoped that a workable neutron based fresh fuel verification scheme can be developed. In order to test the feasibility of such a method, computer based simulations and in field tests were undertaken. This testing procedure, as in the case of gamma based FFV testing, was designed to encompass all conceivable missing fuel combinations. Also, as in the case of gamma based FFV, in field tests were carried out to confirm the accuracy and reliability of the technique.